US-APWR Equipment Environmental Qualification Program

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Abstract

Structures, systems and components (SSC) that are used to construct the US-APWR need to function in all anticipated environments associated with normal and accident conditions. Good engineering practices, as well as adherence to applicable regulations and standards establishes the need to demonstrate that the SSCs important to safety are capable of fulfilling their design functions in design basis environmental conditions, including seismic events. SSCs that are classified as being important to safety are SSCs whose successful operation provides reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
Equipment Qualification is defined as a systematic approach, by one or more defined methods, to demonstrate and document that important to safety or safety-related SSCs are qualified for use in all anticipated environmental conditions. This qualification provides assurance that SSCs can fulfill their intended safety functions during normal, testing, accident and post accident conditions, including postulated design basis accidents. The Design Control Document (DCD) is the basic document for licensing of the US-APWR standard plant for construction and operation within the USA using a combined license (COL) application. The DCD follows the process established by the U.S. Nuclear Regulatory Commission (NRC) pursuant to Title 10, Code of Federal Regulations (CFR), Sections 50.34 and 52.47 (10 CFR 50.34 & 52.47) which requires that the means and methods for Equipment Qualification be properly identified and established. This licensing process requires that the COL applicant identify and establish means and methods for Equipment Qualification.
MHI will supply a standard plant to utilities (licensees) who will obtain a license by submitting a COL application to the NRC using the licensing process delineated in 10 CFR 52.79. The actual design, procurement, construction, testing, turnover and early operation of the facility are collectively referred to as a US-APWR Project. The US-APWR Equipment Qualification Program (EQP) is implemented on a Project basis.

This technical report first identifies the regulatory basis and supporting industry standards applicable to the equipment qualification process, and secondly describes the equipment qualification criteria and methodology.

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List of Acronyms

AISC	American Institute of Steel Construction
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
AWS	American Welding Society
CCASSI	Critical Characteristics for Acceptance of Seismically Sensitive Items
CFR	Code of Federal Regulations
CMTR	Certified Mill Test Report
COL	Combined License
CSDRS	Certified Seismic Design Response Spectra
DBA	Design Basis Accident
DCD	Design Control Document
EMC	Electromagnetic Compatibility
EMI	Electro-Magnetic Interference
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
EQP	Equipment Qualification Program
EQSDS	Equipment Qualification Summary Data Sheet
EQSR	Equipment Qualification Summary Report
FIRS	Foundation Input Response Spectra
GDC	General Design Criteria
GMRS	Ground Motion Response Spectra
HELB	High Energy Line Break
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronic Engineers
ISRS	In-Structure Response Spectra
LOCA	Loss-of-coolant Accident
MHI	Mitsubishi Heavy Industries, Ltd.
MOV	Motor-Operated Valve
MFLB	Main Feedwater Line Break
MSLB	Main Steam Line Break
NIAC	Nuclear Industry Assessment Committee
NRC	Nuclear Regulatory Commission
NUPIC	Nuclear Procurement Issues Committee

OBE	Operational Basis Earthquake
OEQP	Operating Equipment Qualification Program
OL	Operating License
PCCV	Pre-stressed Concrete Containment Vessel
PEQO	Project Equipment Qualification Organization
PEQP	Project Equipment Qualification Program
PS/BS	East/West Power Source Buildings
QA	Quality Assurance
QAP	Quality Assurance Program
RB	Reactor Building
RCS	Reactor Coolant System
RFI	Radio Frequency Interface
RG	Regulatory Guide
RRS	Required Response Spectra
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRSS	Square Root Sum of the square
SSC	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
TID	Technical Information Document
TMI	Three Mile Island
TRS	Test Response Spectra

1.0 PURPOSE

The purpose of this technical report is to describe the Equipment Qualification Program (EQP) applicable to important to safety structures, systems and components used to construct a Mitsubishi Heavy Industries, Ltd. (MHI), US-APWR nuclear power plant.

The purpose of the US-APWR EQP is to both enhance the quality of the US-APWR and to comply with the requirements of Title 10 Energy, Code of Federal Regulations (CFR) and in particular Parts 50.49 and Appendix A of 10 CFR 50, General Design Criteria (GDC) 2 and 4. The Introduction to 10 CFR 50, Appendix A includes the following wording (emphasis added in bold):

> "Under the provisions of 10 CFR 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. Under the provisions of 10 CFR 52.47, 52.79, 52.137, and 52.157, an application for a **design certification**, **combined license**, **design approval, or manufacturing license**, respectively, must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components **important to safety**; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. …"

There are 64 GDCs applicable to power reactors, however, GDCs 1, 2 and 4 are the primary criteria dealing with equipment qualification requirements. GDC No.1 requires quality assurance programs and records applicable to a power reactor project. GDCs 2 and 4 are quoted below.

GDC No. 2 states:

> "*Criterion 2--Design bases for protection against natural phenomena*. **Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.** The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

GDC No. 4 states:

> "*Criterion 4--Environmental and dynamic effects design bases*. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be **compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.** These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic

> effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

These requirements are partially codified in 10 CFR 50.49 which states, in part:

> "Each holder of or an applicant for an operating license issued under this part, or a combined license or manufacturing license issued under part 52 of this chapter, other than a nuclear power plant for which the certifications required under 10 CFR 50.82(a)(1) or 10 CFR 52.110(a)(1) of this chapter have been submitted, **shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.** For a manufacturing license, only electric equipment defined in paragraph (b) which is within the scope of the manufactured reactor must be included in the program."
>
> [Note that American Society of Mechanical Engineers (ASME) QME-1 provides guidance on the qualification of active mechanical components (e.g., valves) and has been endorsed by the NRC in lieu of a direct statutory reference.]

These regulations require that the US-APWR be designed and constructed using structures, systems and components (SSCs) that will withstand both normal (mild) and potentially harsh environments associated with accident conditions. The US-APWR EQP provides a structured approach to complying with these requirements. Equipment Qualification is defined in part as a systematic approach to generating documentation, by one or more defined methods, to demonstrate that important to safety SSCs are qualified for operation in all anticipated environmental conditions. This qualification provides assurance that the SSC can fulfill its intended safety function during normal, testing, accident and post accident conditions, including postulated design basis accidents. Anticipated environmental conditions are the expected temperature, pressure, humidity (including submergence or impingement), chemical, radiation, seismic, aging and synergistic effects that an SSC may experience during normal, accident, testing and post accident conditions at the location within the facility at which the important to safety SSC is installed. Important to safety is an expression for defining the characteristics of an SSC in relation to the SSC's purpose in the plant. Safety-related SSCs are a subset of important to safety SSCs and have a clear definition (10 CFR 50.49) and for electrical equipment are referred to as Class 1E equipment (IEEE Std 323-1974). Safety-related or important to safety active mechanical equipment is addressed by ASME QME-1, "*Qualification of Active Mechanical Equipment Used in Nuclear Power Plants*." Safety-related SSCs are defined as:

> "…those structures, systems and components that are relied upon to remain functional during and following design basis accidents 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 shutdown condition; or
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11 of Title 10 of the Code of Federal Regulations as applicable."

Important to safety SSCs, which must be qualified pursuant to the above requirements, include other plant SSCs that provide a safety function. One definition of important to safety is SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

It is important to note that certain non-safety related SSCs, whose failure under postulated accident conditions could prevent safety-related equipment from accomplishing its safety function, must also be qualified; i.e., are included in the US-APWR EQP. In addition, certain post accident monitoring and related systems as provided in Rev. 4 of Regulatory Guide 1.97, must also be included in the EQP.

Thus, all important to safety SSCs used to construct the US-APWR need to be appropriately qualified. The US-APWR EQP provides a structured approach to accomplish these qualification requirements.

1.1 Applicable Codes and Standards

Questions arise concerning which version of various guidance documents (Regulatory Guides, Industry Standards and Industry Practices) apply to a certain element of the equipment qualification program. There are cases where one version (revision) may be referenced in one area of this technical report and another revision referenced somewhere else. This occurs because the guidance documents are not fully synchronized to the ongoing revision process that occurs with these documents. NUREG 0800, the Standard Review Plan for licensing documentation discusses this and indicates that the licensing documents should reference the versions that are in effect approximately 6 months from the date of submission of a licensing document.

In the case of this technical report, which describes the generic US-APWR EQP, various current guidance documents have endorsed different versions of the same national standard. It should be recognized that these irregularities will be resolved when a specific EQP is established for a specific US-APWR project. For a specific project, the applicable combined license (COL) application will be the controlling licensing document for the project. The implementation of a Project Specific EQP will address these irregularities. The guidance documents are not mandatory, but instead only provide general direction on how statutory requirements are to be met. As such, as long as the established programs address statutory requirements, the NRC has indicated that this is an acceptable methodology. In summary, various revisions of EQ guidance documents may be referenced primarily because that aspect of the EQ program is based on the version of the guidance document presently published by the NRC. As an example, Regulatory Guide 1.89 (rev. 0, 1974 & rev. 1 1984) are directed primarily at qualifying electrical equipment in harsh environments. They both endorse IEEE Std 323-1974. Regulatory Guide 1.209 (rev. 0, 2007), which addresses safety-related computer-based I&C systems, primarily located in mild environments, endorses IEEE Std 323-2003. Similar examples exist with other elements of the EQP.

2.0 SCOPE

This technical report describes the US-APWR EQP. The EQP is described in the US-APWR Design Control Document (DCD). The equipment qualification process is required for the life of the facility (i.e., ~60 years). However, the US-APWR EQP covered by this Technical Report only addresses the period from plant licensing (Combined License [COL]) submittal for a project through initial plant operations. Figure 2.1 illustrates the various phases for EQ and the US-APWR EQP. The roles and responsibilities for an EQP change, depending on the phase of a project being considered. At the DCD phase, Mitsubishi Heavy Industries, Ltd. (MHI) is responsible for establishing a generic EQP. EQ covers:

- Mechanical, Electrical and I&C equipment important to safety
- Seismic qualification of important to safety equipment

Plant piping systems are analyzed under ASME requirements and are, therefore, not directly covered by the EQP (active components such as valves in these piping systems are covered by the EQP).

For each US-APWR project contracted for delivery to a U.S. utility, the Project Equipment Qualification Program shall be established in such a way that it applies to all project activities, including those associated with design, procurement, equipment fabrication, plant construction and plant startup, as well as turnover phases to the utility. At the completion of the Equipment Qualification Program, equipment qualification records will be turned over to the utility as the basis for the utility's operating EQP.

2.1 EQ Program Technical Report Layout

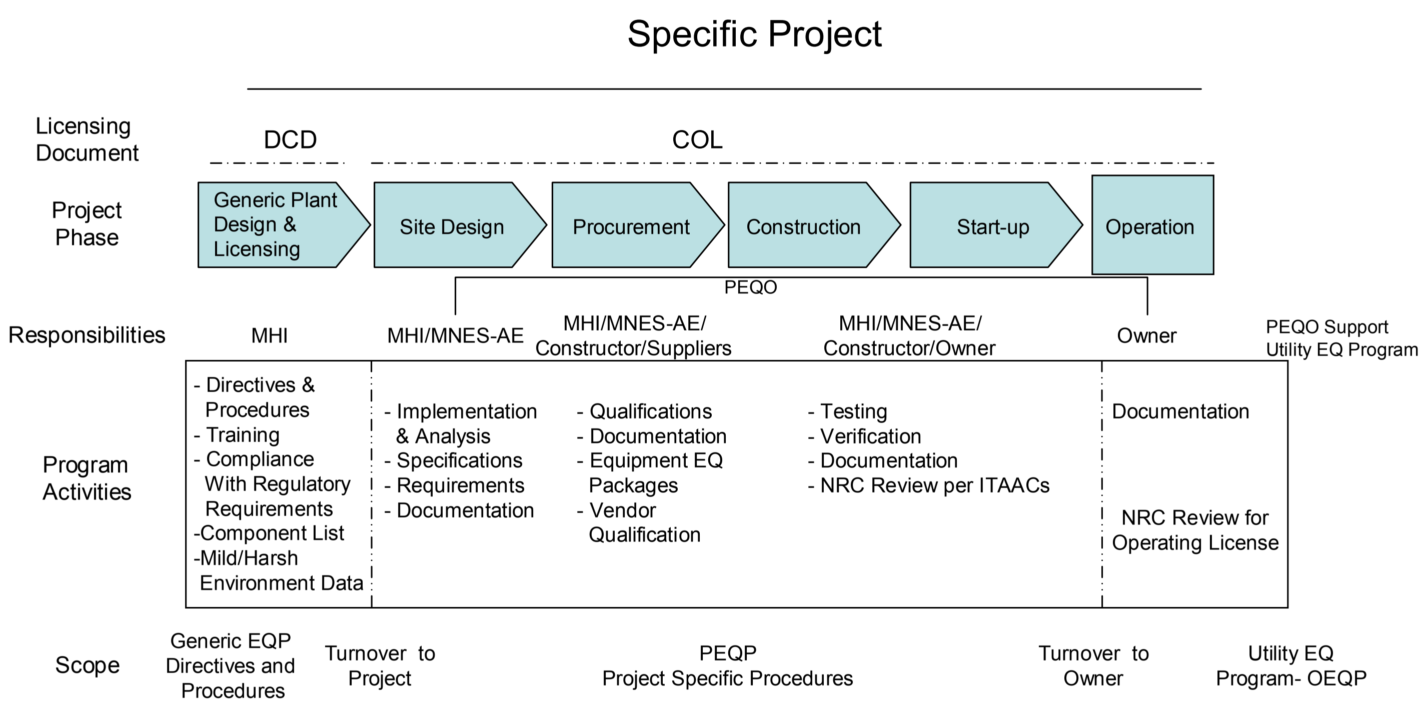
Sections 1.0 and 2.0 of this Report provide the basis for the formal adaptation for an EQP. Section 3.0 describes the applicable statutory (Title 10, Energy Code of Federal Regulation) requirements, the regulatory guidance (Regulatory Guides and NUREGS), industry codes and standards, and industry practices applicable to the US-APWR EQP. Section 4.0 describes the Qualification Criteria for mild and harsh environment definitions, aging, operability time, performance criterion, margin, treatment of failures and traceability. Section 5.0 discusses the Design Specifications for normal and abnormal operating conditions, and design basis accident conditions, and Section 6.0 describes the Qualification Methods including type test, analysis, operating experience, on-going qualification and combination of methods.

Figure 2-1 illustrates the scope of a project-specific EQP. Projects can be divided into phases, and although the distinctions between these phases are in practice not sharp, from a planning or management perspective, they are unique. The DCD process licenses a standardized US-APWR, including the generic EQP. When a plant is sold, an application is normally submitted for a COL that covers the time period associated with site-specific design, procurement, construction and startup phases of the project. Site design phase in COL also includes development of site-specific environmental data that is used to verify the standard design, and evaluate site-specific portions of the plant, including input design parameters for EQ. Following initial power ascension testing, the COL transitions to an operating license (OL). The PEQP covers the time period and phases associated with the COL up to the point when an OL is authorized. At the point that the plant is complete, and the project transitions to the owner and the PEQP transitions to the owner's (plant licensee's) equipment qualification program. The licensee's equipment qualification program is an operating program primarily designed to assure qualified replacement parts are used during the life of the plant. The PEQP is a design, procure, construct and test equipment qualification program. Thus, for each US-APWR, there are three applicable and distinct equipment qualification programs. They are:

1. **Generic Equipment Qualification Program (EQP):** The program that provides the foundation for the project-specific equipment qualification programs. This program and the associated commitments to the NRC are addressed in the DCD and in this Technical Report.
2. **Project Equipment Qualification Program (PEQP):** An EQP that is implemented under, and governed by, the MHI/MNES Equipment Qualification (EQ) Directives and Procedures for a specific project. A PEQP generates and maintains EQ records in accordance with established project program procedures and quality assurance requirements.
3. **Licensee's Operating Equipment Qualification Program (OEQP):** The plant owner's long-term operating equipment qualification program. The OEQP is based on the records and results of the PEQP. The transition from the PEQP to OEQP occurs at the time of initial plant licensure (OL). The plant EQ program covers the life of the plant (~60 yrs) and is discussed in the licensee's COL application.

The Technical Report primarily concerns the generic US-APWR EQP and its implementation as a PEQP. The next section describes the statutory and regulatory basis for EQ.

Figure 2.1 Project EQ Program (PEQP) Implementation Framework



DCD – Design Control Document – Generic to all US-APWRs

COL – Combined License Issued to Plant Owner/Operator – Specific to Each Plant/Site

PEQO – Project Equipment Qualification Organization

PEQP – Project Equipment Qualification Program

OEQP – Operating Equipment Qualification Program

3.0 REGULATORY STATUTES, REGULATORY GUIDES, INDUSTRY CODES and STANDARDS APPLICABLE TO EQUIPMENT QUALIFICATION

The regulatory basis for the US-APWR EQP is briefly described in Section 1.0 of this Technical Report. This section expands on the initial regulatory basis and identifies additional guidance documents applicable to the implementation of the EQP. The requirements and guidance provided in these documents form the basis for the EQ Procedures for the US-APWR EQP as described in the DCD.

This section first identifies the major applicable federal statutes and the associated guidance documents (Regulatory Guides). Regulatory Guides are issued by the NRC as guidance to addressing regulatory requirements. Regulatory Guides usually endorse one or more industry codes and standards. For EQ, these are, for the most part, issued by the Institute of Electrical and Electronic Engineers (IEEE) and the American Society of Mechanical Engineers (ASME). Finally, industry groups such as the Electric Power Research Institute (EPRI), Nuclear Procurement Issues Committee (NUPIC) and Nuclear Industry Assessment Committee (NIAC) provide additional guidance and direction to various elements of an effective EQP.

Attachment A summarizes identified regulations, codes, standards and industry documents applicable to the US-APWR EQP. This section discusses the major statutory (10 CFR), regulatory (Regulatory Guides), standards (industry, e.g., IEEE) and other documents that form the foundation for the US-APWR EQP. There are additional Regulatory Guides, industry codes and standards applicable to certain elements of the EQP that are not listed in this section but are listed in the References section (Section 7.0) of this Technical Report.

3.1 Code of Federal Regulations and General Design Criteria

The design, construction and operation of a power reactor are governed by general requirements, or design criteria, by which each type of power reactor must comply. These general requirements assure that, regardless of reactor type, adherence to the principles of these criteria will result in a facility that minimizes the risk to workers and the public. These regulations are invoked in Title 10, Energy in the Code of Federal Regulations (CFR), Parts 34, 50 and 52, and particularly in 10 CFR 50, Appendix A. Adherence to the General Design Criteria (GDC) contained in Appendix A is a condition of licensure and is, in part, the basis for the need for an equipment qualification program. As such, the GDCs form the basis for standards promulgated by IEEE and ASME pertaining to the EQ. The applicable GDCs, along with a brief explanation, are listed below.

3.1.1 10 CFR 50.49 Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

This is the key statute regarding equipment qualification for important to safety electrical equipment. It should be noted that this statute defines which equipment needs to be qualified and the specifications to which it needs to be qualified. 10 CFR 50.49 requirements are clarified in Regulatory Guide 1.89 and together they reference IEEE Std 323 as an acceptable methodology to follow in qualifying electrical equipment. In 10 CFR 50.49 and IEEE Std 323, a distinction is made between Harsh and Mild environments. In general SSCs, located in harsh environments are qualified pursuant to IEEE Std 323 (and other applicable IEEE standards) while mild environment SSCs can be considered qualified provided the environmental conditions are specified in a purchase specification and the vendor provides equipment meeting these purchase specification requirements. An alternate methodology to qualifying equipment in harsh environments is to follow commercial dedication procedures, where acceptable, as outlined in EPRI and NRC approved EPRI topical reports.

3.1.2 10 CFR 50 Appendix A: General Design Criteria for Nuclear Power Plants

GDC 1: QUALITY STANDARDS AND RECORDS
This GDC requires work that impacts important to safety SSCs be performed and documented using approved procedures and quality standards. This in essence requires a nuclear grade Quality Assurance Program (QAP) that meets the requirements of 10 CFR 50, Appendix B. This in turn implies that all records associated with an EQP be maintained in accordance with the project QAP.

The US-APWR DCD refers to the QAP for the US-APWR as required by GDC 1, quoting the regulation as follows:

"Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit."

The US-APWR EQP encompasses the above criteria.

GDC 2: DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA
This GDC requires important to safety SSCs be qualified to withstand the effects of a seismic event or other adverse natural phenomena (tornadoes, hurricanes, floods, tsunami, and seiches). This GDC applies to both mechanical and electrical important to safety equipment. Important to safety equipment is located within the plant containment as well as other plant buildings. This implies that important to safety equipment must be qualified to withstand anticipated seismic events for a specific plant (project). Thus the seismic qualification requirements are based on both plant-specific seismic criteria as well as the location within the facility.

The US-APWR DCD refers to the design for the US-APWR as required by GDC 2, quoting the regulation as follows:

"Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed."

The US-APWR EQP encompasses the above criteria.

GDC 4: ENVIRONMENTAL AND DYNAMIC EFFECTS DESIGN BASES.
This GDC requires important to safety SSCs be compatible with the environmental effects associated with normal operations (including testing and maintenance) and postulated accident, including post accident/recovery conditions. This GDC also requires this equipment to withstand the effects of high energy line breaks and potential flooding due to line breaks. This implies that safety-related important to safety equipment located within the containment or support structures must be qualified to withstand anticipated conditions associated with a loss-of-coolant accident (harsh environment), unless the location in the plant will not change before or after a postulated accident (e.g., control room, also known as a mild environment).

The US-APWR DCD refers to the design basis of Environmental and Dynamic Effects for the US-APWR as required by GDC 4, quoting the regulation as follows:

"Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

The US-APWR EQP encompasses the above criteria.

GDC 19 CONTROL ROOM
This GDC requires the control room design to allow safe operation during normal and accident (including post accident) conditions and to prevent radiation exposure (above a certain threshold) to the control room or personnel within it. Control room equipment must therefore be qualified for operation for both seismic and environmental conditions. The US-APWR EQP includes the safety-related plant instrument and control systems including computer-based systems.

The US-APWR DCD refers to the design of the control room for the US-APWR as required by GDC 19, quoting the regulation as follows:

"A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under 10 CFR 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident."

The US-APWR EQP encompasses the above criteria.

GDC 20 PROTECTION SYSTEM FUNCTIONS
This GDC requires the plant protection systems operate automatically during all anticipated plant conditions to protect the plant and fuel. This in turn implies that the plant protection systems be designed for all anticipated environmental and seismic conditions, and as such, the protection system components must be qualified.

The US-APWR DCD refers to the design of the protection systems for the US-APWR as required by GDC 20, quoting the regulation as follows:

"The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety."

The US-APWR EQP encompasses the above criteria.

GDC 21 PROTECTION SYSTEM RELIABILITY AND TESTABILITY
This GDC requires the plant protection system to be designed for high reliability and testability. This in turn implies that the plant protection systems be designed for all anticipated environmental and seismic conditions as was stated for GDC 20. This GDC addresses automated and manual testing and redundancy of plant protection systems under operating conditions. Thus, the protection system design criteria for EQ should include accomplishing these requirements under the environmental parameters for its location. This would indicate that testing a part of the protection system would not alter the EQ environment protection for the remainder of the protection system. An example of this would be if a protection system instrument cabinet had to be opened to perform a test thus exposing the remainder of the protection system to an environment outside of the EQ qualification.

The US-APWR DCD refers to the design of the protection systems for the US-APWR as required by GDC 21, quoting the regulation as follows:

"The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred."

The US-APWR EQP encompasses the above criteria.

GDC 23 PROTECTION SYSTEM FAILURE MODES
This GDC requires that the protection system failure modes address environmental extremes (e.g., freezing) so that the equipment fails in a safe manner. The EQP must therefore identify possible extreme environmental conditions.

The US-APWR DCD refers to the design of the protection systems for the US-APWR as required by GDC 23, quoting the regulation as follows:

"The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced."

The US-APWR EQP encompasses the above criteria.

GDC 24 SEPARATION OF PROTECTION AND CONTROL SYSTEMS
This GDC is related to GDC 19 and 20, and therefore, aspects of the EQP are invoked. This GDC addresses that protection system shall be separated from non-safety control systems. Also, failure of non-safety control systems does not allow affecting the protection system. The EQP aspects of this GDC are in the physical isolation of the various components, systems and structures of the protection system.

The US-APWR DCD refers to the design of the protection systems for the US-APWR as required by GDC 24, quoting the regulation as follows:

"The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

The US-APWR EQP encompasses the above criteria.

GDC 29 PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES
This GDC is related to GDC 19 and 20, and therefore, aspects of the EQP are invoked. This GDC addresses the normally expected events that may occur during power operation. This could include events that would have an outside environmental cause. These environmental parameters can be plant-specific and are described in general in Section 4.0 of this Report.

The US-APWR DCD refers to the design of the protection systems for the US-APWR as required by GDC 29, quoting the regulation as follows:

“The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.”

The US-APWR EQP encompasses the above criteria.

Other GDC’s (e.g., 14, 22, 30 and 60) have elements of EQ within them but are not as prominent as these cited in this Section. Section 4.0 of this Technical Report provides additional discussions on GDC 14, 22, and 30.

3.2 NRC Staff Requirements Memoranda

10 CFR 50.34 (f) Post-TMI Requirements. These memoranda primarily concern the need for post-accident monitoring and recovery equipment to be qualified for post-accident conditions.

3.3 NRC Regulatory Guides

Regulatory Guides are issued to clarify statutory requirements and to advise licensees of the NRC’s acceptance of using certain professional society codes and standards to meet these requirements. Regulatory Guides will often point out differences acceptable to the staff in certain aspects of the codes and standards that the NRC believes should be followed to meet statutory requirements. The Regulatory Guides (RG) listed below apply to the US-APWR EQP in the areas identified.

RG 1.22, “PERIODIC TESTING OF PROTECTION SYSTEM ACTUATION FUNCTIONS”

This Regulatory Guide addresses the requirements for periodic automatic and manual tests.

The US-APWR EQP function encompasses performance of the automatic and manual tests. See Section 3.1, (1), GDC 21 for additional information

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.29, REVISION 4, “SEISMIC DESIGN CLASSIFICATION”
In some cases, a Seismic Category I (safety-related) SSC will also include Seismic Category II (Non-Safety Related) equipment. However, Seismic Category II equipment is designed so that a Safe Shutdown Earthquake (SSE) will not result in a failure that would impede the Category I SSC from performing its safety function.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.40, “QUALIFICATION TESTS OF CONTINUOUS-DUTY MOTORS INSTALLED INSIDE THE CONTAINMENT OF WATER-COOLED NUCLEAR POWER PLANTS”
IEEE Std 334-1971, "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," was prepared by Subcommittee 2 of the Joint Committee on Nuclear Power Standards of the Institute of Electrical and Electronics Engineers, Inc. (IEEE), and was subsequently approved by the IEEE Standards Committee on September 16, 1971.

The Standard delineates specific procedures for the qualification testing of Class I motors to demonstrate adequacy of design for service within the containments of nuclear power plants. These procedures provide for testing under conditions simulating those imposed during normal operation in addition to those resulting from a design basis loss-of-coolant accident.

The Standard specifies procedures for accomplishing accelerated aging of components to simulate the effects of long-term operation, including radiation effects, and for subjecting a prototype aged motor to combined (steam) pressure, temperature, and chemical environments approximating those of the design basis loss-of-coolant accident.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.63, REVISION 3, “ELECTRIC PENETRATION ASSEMBLIES IN CONTAINMENT STRUCTURES FOR LIGHT WATER-COOLED NUCLEAR POWER PLANTS”

IEEE Std 317-1983, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," was prepared by a working group of Subcommittee 1, General Plant Criteria, of the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers (IEEE) and was subsequently approved by the IEEE Standards Board on September 23, 1982. This standard prescribes requirements for the design, construction, testing, qualification, and installation of electric penetration assemblies in containment structures for stationary nuclear power generating stations.

Section 6.2.8(5) of IEEE Std 317-1983 requires that the duration of maximum short circuit current flow in test specimens of electric penetration assemblies be no less than 0.033 second.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.73, REVISION 0, JANUARY 1974, “QUALIFICATION TESTS OF ELECTRIC VALVE OPERATORS INSTALLED INSIDE THE CONTAINMENT OF NUCLEAR POWER PLANTS”
Provides guidance for electric valve operators installed inside the Containment of Nuclear Power Plants.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.89, REVISION 1, “ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS”

This Regulatory Guide was issued in 1974 and was revised in 1984. The DCD and SRP 0800 reference the 1974 version. However, guidance provided in the 1984 version will be considered in the EQP. The 1984 version endorses IEEE Std 323-1974 just as the 1974 version did. Regulatory Guide 1.89 provides the principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. However, certain NUREG-0588 Category I guidance may be used where relevant guidance is not provided in Regulatory Guide 1.89. The 1984 version provided additional guidance in addressing items associated with environmental conditions caused by chemicals, radiation, high energy line breaks, pressure changes, humidity and synergistic effects. This guidance is incorporated in the EQP. A major characteristic of equipment important to safety under the US-APWR EQP is environmental and dynamic qualification of all such equipment as addressed by Reg. Guide 1.89 as follows:

Section 50.49 requires that three categories of electric equipment important to safety be qualified for their application and specified performance and provides requirements for establishing environmental qualification methods and qualification parameters. These three categories are (1) safety-related electric equipment (Class 1E), (2) non-safety related electric equipment (non-Class 1E) whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by safety-related equipment, and (3) certain post accident monitoring equipment. This regulatory guide applies only to these three categories of electric equipment important to safety.

This Regulatory Guide describes a methodology acceptable to the NRC staff for complying with Section 50.49 of 10 CFR 50 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

The environmental qualification of the equipment covered by this Regulatory Guide is by an appropriate combination of testing and analysis. This Regulatory Guide addresses equipment that is located in both harsh and mild environments. Equipment that is located in harsh environments is qualified following the guidance provided in this Regulatory Guide and IEEE Std 323, and other documents as listed in this section of the Report. Mild environments include those that are not adversely affected by plant accidents. Therefore, qualification of equipment in mild environments for temperature, humidity and radiation is normally done by analysis of component vendor specifications, room ambient conditions and heat rise calculations for the installed configuration. The vendors will normally test this equipment (in most cases using nationally recognized testing agencies such as Underwriters Laboratory) and certify its use in mild environments by industry recognized ratings (e.g., NEMA 1 – indoor locations, NEMA 3R outdoor subject to rain). As part of the vendor qualification, the vendor's test programs and manufacturing processes are audited. Seismic qualification and Electro-Magnetic Interference (EMI) qualification are normally done by type testing (for seismic, see RG 1.100 and IEEE Std 344, and for EMI, see RG 1.180). This type of equipment generally has no known aging failure mechanisms and usually has an expected long service life. However, random failures and other long term issues are normally detected by an operating plant's periodic surveillance, calibration and testing programs. These types of failures, if they were to occur, are anticipated by the defenses in depth approach (i.e., single failure criteria) of multiple safety system divisions and associated separation criteria. Other environmental qualification requirements (e.g., synergistic effects) for equipment in mild environments are normally covered by analysis or various forms of vendor certifications.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.97, REVISION 4, "CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION FOR NUCLEAR POWER PLANTS" - ENDORSES IEEE STD 497-2002
The equipment regulated by this RG is used to process and display signals from accident monitoring instrumentation of all variable types.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.100, REVISION 3, "SEISMIC QUALIFICATION OF ELECTRIC AND MECHANICAL EQUIPMENT FOR NUCLEAR POWER PLANTS"
This regulatory guide endorses IEEE Std 344-2004, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The Nuclear safety-related equipment governed by the US-APWR EQP is designated Seismic Category I. It is designed and qualified to withstand the cumulative effects of a minimum of five (5) Operational Basis Earthquakes (OBEs) and one (1) SSE without loss of safety function or physical integrity. The input spectrum is selected to envelope all anticipated applications. Conformance to this envelope for specific applications is discussed in DCD Section 3.10.1.1.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

DG-1175 - The NRC has indicated that the Regulatory Guide will be formally issued on or about December 2008 and will endorse IEEE Std 344-2004, "*Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*", and ASME QME-1 2007, "*Qualification of Active Mechanical Equipment in Nuclear Power Plants,*" as the basis for seismic qualification of important to safety mechanical and electrical equipment. QME-1 provides additional guidance on qualification of mechanical active components such as valves, pumps and non-metallic parts.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

ASME Code generally invokes American Institute of Steel Construction (AISC) for support design. However, for non-ASME supports and structures, such as those for electrical equipment, AISC N690 Code for safety-related steel structures is applicable. The US-APWR is committed to AISC N690, including Supplement 2, and specifies to apply N690 stress coefficients to the allowable stresses of AISC and AISI in order to determine allowable stresses to be used.

RG 1.131, REVISION 0, AUGUST 1977, "QUALIFICATION TESTS OF ELECTRIC CABLES, FIELD SPLICES, AND CONNECTIONS FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS"

This regulatory guide endorses IEEE Std 383, "IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations," 2003.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.156, "ENVIRONMENTAL QUALIFICATION OF CONNECTION ASSEMBLIES FOR NUCLEARPOWER PLANTS"

This regulatory guide endorses IEEE 572-1985. The US-APWR equipment qualification program employs the recommendations of Regulatory Guide 1.156 in specifying the qualification program plans where this guide supplements the guidance of IEEE 572 to demonstrate conformance with the guidance of IEEE 323.

RG 1.158, "QUALIFICATION OF SAFETY-RELATED LEAD STORAGE BATTERIES FOR NUCLEAR POWER PLANTS"

This regulatory guide endorses IEEE 535-1986. The US-APWR equipment qualification program employs the recommendations of Regulatory Guide 1.158 in specifying the qualification program plans where this guide supplements the guidance of IEEE 535 to demonstrate conformance with the guidance of IEEE 323.

RG 1.180, REVISION 1, 2003, “GUIDELINES FOR EVALUATING ELECTROMAGNETIC AND RADIO-FREQUENCY INTERFERENCE IN SAFETY-RELATED INSTRUMENTATION AND CONTROL SYSTEMS”

This Regulatory Guide endorses Military Standard MIL-STD-461E and the International Electrotechnical Commission (IEC) 61000 series of EMI/ Radio Frequency Interface (RFI) test methods. Section 50.55a(h) of 10 CFR Part 50 requires that protection systems meet the requirements of the Institute of Electrical and Electronics Engineers (IEEE) standard (Std) IEEE Std 603-1991, “Criteria for Safety Systems for Nuclear Power Generating Stations.” 10 CFR 52.47(a)(vi) requires that an application for design certification must state the tests, inspections, analyses, and acceptance criteria that are necessary and sufficient to provide reasonable assurance that a plant will operate within the design certification. Methods for addressing electromagnetic compatibility (EMC) constitute Tier 2 information under the 10 CFR Part 52 requirements. The Regulatory Guide also endorses design and installation practices described in IEEE Std 1050-1996, “IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations.”

Safety-related electronic equipment subject to potential EMI/RFI caused malfunctions is evaluated in the EQP to verify acceptable operation using the guidance provided in this Regulatory Guide.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.209 (FORMERLY DG 1077), “GUIDELINES FOR ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED MICROPROCESSOR-BASED EQUIPMENT IMPORTANT TO SAFETY IN NUCLEAR POWER PLANTS” - ENDORSES IEEE STD 323-2003

This referenced Equipment, which consists of safety-related computer-based I&C systems, is located in a mild environment. There is no change in the environment due to plant accidents. This equipment is tested and analyzed to satisfy the mild environmental qualification requirements.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

RG 1.183 (FORMERLY DG 1081), "ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS"
This Regulatory Guide provides the basis for the radiation source term for evaluating Design Basis Accidents and the resulting radiation doses input into the EQP. This input is a critical characteristic of equipment important to safety and these radiation doses are to be used in the purchase specification for this equipment. With respect to the Regulatory Guide, it provides the basis for the radiation source term as found in the introduction to the guide:

This guide provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information," requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, "Reactor Site Criteria," Section 100.11 "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, "Clarification of TMI Action Plan Requirements." The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility Final Safety Analysis Report (FSAR). Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis.

The US-APWR EQP encompasses the requirements specified in this Regulatory Guide.

3.4 ASME and Other Industry Standards Codes

The following ASME codes and industry codes are applicable to the US-APWR during the design, procurement, and construction phases.

ASME Section III addresses mechanical components and systems important to safety and the qualification needed to meet the mechanical requirements of that code. This also includes the material specifications of those mechanical components and systems important to safety especially for the piping systems and components, including pipe, valves and other fittings that make up these systems. Seismic analysis criteria and methods for the mechanical and piping systems for the US-APWR follow this code. These requirements extend to the support systems and structures for the mechanical components and systems important to safety. Individual portions of the support structures are qualified by referring to the American Institute of Steel Construction (AISC) manual and accompanying analysis for steel structures (see below).

Active support components such as springs and snubbers are also covered under ASME codes. The EQ requirements for ASME extend beyond confirmation of appropriate material specification via the Certified Mill Test Report (CMTR) or Certificate of Compliance in the case of items such as valves, pipe support spring hangers, or snubbers. This includes pressure testing of valves, chemical and environmental qualification for the material selection, painting of components, and other EQ factors, as each design requires.

ASME NQA-1 provides the Quality Assurance requirements needed for the above mechanical systems and components. ASME NQA-1 also includes the documentation requirements necessary to meet this code.

ASME QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, 2007, provides guidance on qualifying active mechanical equipment.

The American Institute of Steel Construction (AISC) manual and material specifications are used in design, engineering and fabrication of individual portions of the support structures. This includes fasteners, steel shapes, and welding specifications invoked from the American Welding Society (AWS) D1.1 code for prequalified welds. These prequalified welds of steel structures allow for support fabrication and are from the AWS code.

AWS standards are used for welding of metallic components under both ASME and AISC codes and will be addressed on a project-specific basis to allow inclusion of site-specific welding procedures at the time of construction for a US-APWR project.

See References section for additional Industry Codes applicable to the US-APWR EQP.

The US-APWR EQP encompasses the above codes and standards.

3.5 NUREG-Series Publications (NRC Guidance Document for SRP for DCD)

NUREG 0800 addresses the preparation of the US-APWR DCD and an associated COLA FSAR, and the licensing requirements therein. NUREG 0800 Standard Review Plan (SRP) Section 3.10 covers “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment”. Section 3.11 covers “Environmental Qualification of Mechanical and Electrical Equipment.” Refer to the US-APWR DCD and project COLA FSAR for the response to NUREG 0800, Section 3.10 and 3.11. The US-APWR DCD can be found as a public document filed with the U.S. Nuclear Regulatory Commission. It provides the licensing basis and plant description for the standard US-APWR. The DCD and its Appendix 3D includes engineering and design details for the components and systems important to safety and added details for the EQP. Appendix 3D is the listing of the equipment and environmental requirements for the components and systems important to safety for the standard US-APWR. The COLA FSAR for a specific US-APWR plant project would include any site-specific items to be included in its EQP.

3.6 IEEE and Other Standards

IEEE Std 344-1987, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"

IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" (see discussion under RG 1.189 for IEEE Std 323-1974).

For additional standards, see Section 7.0, References.

3.7 NSSS Industry Standards

EQ activities are addressed by other NSSS industry organizations, including the Electric Power Research Institute (EPRI), the Nuclear Procurement Issues Committee (NUPIC) and NIAC Nuclear Industry Assessment Committee. The following is a list of EQ relevant documents issued by these organizations.

<u>EPRI</u>

EPRI Commercial Grade Dedication methodologies, as approved by the NRC, are encompassed in the US-APWR EQP. The following EPRI documents address commercial dedication.

EPRI NP 5652 Guideline for the Utilization of Commercial Grade Items in Nuclear Safety-related Applications (NCIG-07), 1988

EPRI TR-102260 Supplemental Guidance for the Application of EPRI NP5652 on the Utilization of Commercial Grade Items, 1994

EPRI TR-106439 Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications, 1996

EPRI TR-1001452 Generic Qualification of Commercial Grade Digital Devices, 2001

EPRI TR-1001468 Generic Qualification of Rosemount 3051N Pressure Transmitter, 2001

EPRI TR-112579 Critical Characteristics for Acceptance of Seismically Sensitive Items (CCASSI), 2000

EPRI TR-1003105 Dedicating Commercial-Grade Items Procured From ISO 9000 Suppliers, 2001

<u>NUPIC</u>

NUPIC Commercial Dedication methodologies, as approved by the NRC, are encompassed in the US-APWR EQP.

NUPIC, Document No. 10 Commercial Grade Survey Description, 1999

NAIC

Audit Procedures and Guidelines available to members only.

4.0 Qualification Criteria

The environmental requirements considered in the design of safety-related equipment are embodied in GDC 2, "Design Bases for Protection Against Natural Phenomena"; GDC 4, "Environmental and Missile Design Bases"; and GDC 23, "Protection System Failure Modes." GDC 1, "Quality Standards and Records," and Criterion III, "Design Control," Criterion XI, "Test Control," and Criterion XVII, "Quality Assurance Record" of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to require that the environmental design of safety-related equipment is verified, documented, and controlled.

The qualification methods described in this Technical Report are used to verify the environmental design basis and capability of the safety-related electrical and mechanical equipment supplied for the US-APWR. Design control, test control, and quality assurance record keeping is performed through the US-APWR Quality Assurance Program. (See Chapter 17 of DCD.)

4.1 Definition of Plant Locations by Type of Possible Environmental Condition

4.1.1 Mild Environment

A mild environment is one that would, at no time, be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. From IEEE, 100 *The Authoritative Dictionary of IEEE Standard Terms*, the definition is: An environment expected as a result of normal service conditions and extremes (abnormal) in service conditions where seismic is the only design basis accident (DBA) of consequence. Typically a mild environment conforms with the environmental parameter limits of Table 4-1.

Mild environments can have exposure to radiation levels during normal operation. Systems important to safety, but not in the containment or other location where they could see the harsh environmental condition described below, would fall into the mild category. These important to safety systems would be evaluated for accident conditions to assure the mild category still applies. Refer to Reg. Guide 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants” for guidance for safety-related computer-based I&C system on this situation. Mild areas are further defined in the US-APWR DCD.

For electrical and mechanical equipment located in a mild environment, acceptable environmental design can be demonstrated by the "design/purchase" specification process for the equipment. The "design/purchase” specification contains a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. The maintenance/surveillance program, in conjunction with the preventive maintenance program, provides assurance that equipment meeting the design/purchase specifications is qualified for the designed life of the component. Compliance by the Licensee (owner) with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," and associated guidance in RG 1.160 are considered sufficient to provide reasonable assurance that environmental considerations established during design are reviewed every refueling outage and maintained on a continuing basis to ensure that the qualified design life has not been reduced by thermal, radiation, and/or cyclic degradation resulting from unanticipated operational occurrences or service conditions.

4.1.2 Harsh Environment

A harsh environment is expected as a result of the postulated service conditions appropriate for the design basis and post-design basis accidents of the station. (A design basis accident is that subset of a design basis accident which requires safety function performance). Harsh environments are the result of a loss-of-coolant accident (LOCA)/high energy line break (HELB) inside containment and post-LOCA or HELB outside containment (this definition from IEEE, *The Authoritative Dictionary of IEEE 100 Standard Terms*).

These special conditions can cause the local environment for the equipment important to safety to be harsh in one or more parameters. These special conditions can result from a DBA, main steam line break (MSLB), main feedwater line break (MFLB), or other high energy line break. High radiation areas outside of the containment are also in a harsh environment.

Equipment that must withstand the environmental conditions that would exist before, during, and following a DBA is qualified for use in harsh environments. A DBA, such as LOCA could subject this equipment to elevated pressures, temperatures, humidity, radiation, and chemical effects (including post accident pH control). This equipment must operate without a loss of its safety function, for the time required to perform its engineered safeguards function(s). These environmental conditions for which the equipment is qualified include applicable time dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and those synergistic effects that have a significant effect on the equipment performance. Equipment identified as being qualified for harsh environment includes the following:

a. Equipment located within containment
b. Equipment subject to HELBs (e.g., MSLB) both inside and outside of containment
c. Other SSCs that connect, support, tie into, or that can influence the equipment listed in “a” and “b” above.

4.2 Aging

Per 10 CFR 50.49 (d) (5), “Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition.” This regulation describes the considerations for the aging testing including preconditioning a given SSC before any further aging tests. This testing is used to help determine the service life of an SSC important to safety. Aging requirements are SSC specific and are implemented on a project specific basis.

4.3 Operability Time

Equipment operating times are determined by the individual piece of important to safety equipment’s function, location, and safety class. This information is the result of engineering analysis for each piece of equipment. This equipment is evaluated for a time dependent safety function after a DBA. The time dependent safety functions are for tripping the reactor after a LOCA or other accident signal, engineered safeguards initiation, post-accident monitoring, or containment isolation. The results are tabulated in the US-APWR DCD.

4.3.1 Shorter Operability Times

Equipment that performs its safety function prior to significant changes in its environment may be qualified for shorter durations. Per Regulatory Guide 1.89, justification for shorter duration includes:

- The consideration of a spectrum of pipe break sizes.
- The potential need for the equipment later in an accident or during recovery operations.

Subsequent failure of the equipment is shown to not be detrimental to plant safety or to mislead the operator.

The determination that the margin applied to the minimum operability time, when combined with other test margins, accounts for uncertainties associated with the use of analytical techniques used to derive environmental parameters, the number of units tested, the production tolerances, and the test equipment inaccuracies.

4.4 Performance Criterion

The qualification test program demonstrates the capability of the equipment to meet the safety related performance requirements. The primary objective of qualification is to demonstrate that equipment, for which a qualified life or condition has been established, can perform its safety functions without experiencing common-cause failures before, during, and after applicable DBAs. The continued capability for this equipment and its interfaces to meet or exceed its specification requirements is provided through a program that includes, but is not limited to, design control, quality control, qualification, installation, maintenance, periodic testing, and surveillance.

4.5 Margin

Environmental parameters applicable to safety-related or important to safety SSCs are listed in engineering specifications prepared in support of the procurement process. These specifications are used in the procurement process by the Project Equipment Qualification Organization (PEQO). In the bid evaluation process for safety-related or equipment important to safety, one of the steps is an engineering evaluation of the equipment’s compliance with its EQ parameters listed in the specification. The term “margin” refers to the extent by which this equipment meets and exceeds the required EQ parameter values. In essence, 10 CFR 50.49(e)(8) states:

> 8) *Margins*. Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

Thus selected equipment will have the qualified margins in EQ parameters to assure that there is adequate conservatism in the equipment. It is not possible to quantify the amount of margin for a given environmental parameter with exact certainty, since in most cases these are commercial transactions and the selected equipment will be evaluated by the PEQO to verify that the margins are acceptable. This means that the procurement process will evaluate the margins and coordinate with the PEQO where appropriate. Table 4-2 lists margin requirements applied.

4.5.1 Normal and Abnormal Extremes

As indicated in Section 7 of IEEE 323, the application of margin is directed at specifying adequate qualification requirements for the most severe service conditions represented by the design basis accidents (that is, high-energy line break accidents and seismic events). Consequently, the US-APWR equipment qualification methodology does not apply any systematic margin to the normal and abnormal environment parameters in defining the qualification conditions.

For electronic equipment not required to operate in a high-energy line break environment, additional margin is included by requiring that the equipment operate through the conservative normal and abnormal service conditions. The environmental parameters at least equal the specified range of service condition parameters. An exception occurs for transmitters where performance verification is completed at 130°F on each transmitter to encompass the specified maximum abnormal conditions. For equipment to be qualified to operate in a high-energy line break environment, qualification to the severe high-energy line break conditions demonstrates ample margin for acceptable performance under certain specified normal and abnormal service conditions.

4.5.2 Aging

No specific margin is applied to the time component in deriving appropriate aging parameters, if margin is included in deriving the accelerated aging parameters employed for simulating each applicable aging mechanism.

Margin may be addressed by demonstrating the adequacy of the aging simulated by test through the calculation of time-temperature equivalence or the comparison of simulated parameters with those applicable to the intended service of the equipment. The installed life of equipment must not exceed the thermal qualified life demonstrated by this calculation. Additionally, the selection and use of the thermal aging parameters both for test and subsequent calculations are subject to criteria, including the following:

- Test temperature must endure for at least 100 hours
- Test temperature must exceed any application temperature (that is, the normal or abnormal environment in which the equipment is to be used, and for which the life is calculated)
- Test temperature must be less than state-change temperature for materials critical to the equipment safety-related function or capability to endure the subsequent design basis accident testing
- A conservative activation energy is used. Activation energies for materials critical to the equipment safety-related function or capability to endure the subsequent design basis accident testing are considered. Materials may have several activation energies, each for a different material property. Relevant material properties are considered.

If margin is not demonstrated through conservatism in the aging parameters or calculation, then a +10 percent time margin is included.

A margin of 10 percent in the other parameters (for example, irradiation, operational cycling) applies to both the aging simulation and the post-accident simulated aging, with few exceptions.

For equipment required by design to perform its safety-related function within a short time period into the design basis accident (that is, within seconds or minutes), and having completed its function, subsequent failure is shown not to be detrimental to plant safety, margin by percentage of additional time or equivalent time-temperature is not applied. Margins for trip function requirements are contained in the worst-case high-energy line break envelope. Test parameters are simulated on a real-time basis with the transient condition margins. Trip signals, once generated by the sensors, are locked in by the protection system and do not reset in the event of subsequent sensor failure.

4.5.3 Radiation

An additional 10 percent is added to the calculated accident dose in specifying the test requirements.

4.5.4 Seismic Conditions

Required response spectra as discussed in Attachment B are the conditions to be enveloped. No amplitude margin is added to these conditions. Peak broadening is also discussed Attachment B. Seismic qualification by analysis addresses margin requirements by other methods of conservatism while using the same sets of requirements - no amplitude margin is included. For qualification tests, the test facility increases the amplitude of seismic profiles by 10 percent to incorporate margin.

For most applications, considerable margin exists with respect to the acceleration levels employed and the width of the response spectra. Further details are addressed in Attachment B.

4.5.5 High-Energy Line Break Conditions

The environmental requirements for equipment specified for high-energy line breaks are selected to by transients resulting from various cases of loss-of-coolant accidents and other high-energy line breaks. These requirements are presented as combined envelopes of pressure or temperature versus time for such regions which includes the high-energy lines and the equipments important to safety. Thus, the envelopes initially contain significant margin with respect to any transient corresponding to a single break.

The US-APWR equipment qualification requires that the envelopes should be additionally considered with a margin of 15°F of the temperature and 10 % of the containment or building design pressure according to IEEE 323-1974.

4.6 Treatment of Failures

The primary purpose of equipment qualification is to reduce the potential for common mode failures due to anticipated environmental and seismic conditions. The redundancy, diversity, and periodic testing of nuclear power plant safety-related equipment are designed to accommodate random failures of individual components.

Where an adequate test sample is available, the failure of one component or device together with a successful test of two identical components or devices indicates a random failure mechanism, subject to an investigation concluding that the observed failure is not common mode. Where insufficient test samples prevent such a conclusion, any failures are investigated to ascertain whether the failure mechanism is of common mode origin. Should a common mode failure mechanism be identified as causing the failure, either a design change is implemented to eliminate the problem or a repeat test completed to demonstrate compliance with the criteria.

For those mild environment equipment items that, through a review of available documentation, are subject to failure during a seismic event due to significant aging mechanisms, the material or component is replaced or monitored through a maintenance/surveillance program.

4.7 Traceability

A system of baseline design documentation is instituted to control the design, procurement, and manufacturing of safety-related products. As part of this quality control program, critical parts are identified and assigned a level of control to reflect the estimate of potential qualification or procurement problems. In addition, levels of quality inspection are also assigned to each part. The baseline design documentation describes the equipment in sufficient detail (drawing number, part number, manufacturer) to establish traceability between equipment shipped and equipment tested in the qualification program.

4.7.1 Auditable Link Document

The purchaser of equipment referencing this program requires an auditable link document that provides a tie between the specific equipment and documentation of qualification reviewed for acceptance under this program. This auditable link document includes one or more of the following sections, as applicable.

4.7.1.1 Equipment Link

This documentation certifies that the plant specific equipment is covered by the applicable equipment test reports. This link reflects a comparison of the as-built drawings, baseline design document or other documentation of the tested equipment to the specific equipment.

4.7.1.2 Component Link

This documentation certifies that the components (for example, replacement parts) used in the specific equipment are represented in the applicable test reports or via analysis under a component aging program. This link applies only to equipment whose documentation references a component testing program. This link reflects a comparison of the as-built drawings, baseline design document, or other documentation of the specific equipment to the component program listing.

4.7.1.3 Material Link

This documentation certifies that the materials used in the equipment are represented in a materials aging analysis. This link applies only to equipment whose documentation references the materials aging analysis and reflects a comparison of the as-built drawings, baseline design document, or other documentation of the plant specific equipment to the materials aging analysis listing.

4.7.2 Similarity

Where differences exist between items of equipment, analysis may be employed to demonstrate that the test results obtained for one piece of equipment are applicable to a similar piece of equipment. Documentation of this analysis conforms with guidelines in IEEE 323 and 627, and subsection 6.2.1.

Table 4-1 Typical Mild Environmental Parameter Limits

Parameter	Limit	Notes
Temperature	≤ 120 ºF ≤ 150 ºF ≤ 130 ºF	Inside Containment（Normal） Inside Containment（Abnormal） Outside Containment
Pressure	Atmospheric	Nominal
Humidity	Non-condensing	
Radiation	$\leq 10^3$ rads gamma $\leq 10^4$ rads gamma	Electronic devices and components Non-electronic devices and components
Chemistry	None	

Table 4-2 US-APWR EQP Margin Requirements

Condition	Parameter	Required Margin	Notes
NORMAL:	Aging	+10%	+10% time margin, +10% radiation and/or selection of conservative test parameters. Comply with guidance of subsection 4.5.2.
ABNORMAL:	Temperature/ Humidity		Margin is in "time" at abnormal test extremes.
	Pressure	None	Nominally atmospheric.
	Radiation	+10%	Include in aging doses, if applicable.
	Chemical Effects	+10%	In alkalinity of adjusted sump pH. Not applicable outside containment.
	Voltage & Frequency	+/- 10%	Simulated during temperature/humidity test.
	Submergence	Note 1	Generally, precluded by design.
ACCIDENT:	Transient Temperature and Pressure		Temperature (+15°F) and pressure (+10 psig peak) margins added to transient profile.
	Chemical Effects	+10%	In alkalinity of adjusted sump pH. Not applicable outside containment.
	Radiation	+10%	Added to calculated total integrated dose.
	Submergence	Note 1	Generally, precluded by design.
	Seismic/ Vibration	+10%	Of acceleration at equipment mounting point for either SSE or line-mounted equipment vibration. (See subsection 4.5.4.)
	Post-accident Aging	+10%	In time demonstrated via Arrhenius time/temperature relationship calculation.

Note:
1. Margin in submergence conditions is achieved by increases in temperature (+15°F), pressure (+10%), and chemistry (+10% in alkalinity of adjusted sump pH). Also, accident conditions submergence testing envelops abnormal conditions submergence conditions.

5.0 Design Specifications

The conditions and parameters considered in the environmental and seismic qualification of US-APWR safety-related or important to safety equipment are separated into three categories: normal, abnormal, and design basis accident. Normal conditions are those sets and ranges of plant conditions that are expected to occur regularly and for which plant equipment is expected to perform its safety-related function, as required, on a continuous, steady-state basis. Abnormal conditions refer to the extreme ranges of normal plant conditions for which the equipment is designed to operate for a period of time without any special calibration or maintenance effort. Design basis accident conditions refer to environmental parameters to which the equipment may be subjected without impairment of its defined operating characteristics for those conditions.

The following subsections define the basis for the normal, abnormal, design basis accident, and post-design basis accident environmental conditions specified for the qualification of safety-related or important to safety equipment in the US-APWR equipment qualification program.

5.1 Normal Operating Conditions

5.1.1 Pressure, Temperature, Humidity and Radiation

The calculated values for temperature, pressure, humidity and radiation during normal operation are specified in Table 5-1 as a function of in-plant location.

5.1.2 Radiation Dose

The radiation environment for qualification of safety-related or important to safety equipment is the Total Integrated Dose resulting from the normally expected radiation environment over the installed life of the equipment, plus that associated with the most severe design basis accident for which the equipment must remain functional. Additionally, dose rates may be a design consideration.

The normal operating dose rates and consequent 60-year design expected doses at various locations inside containment are derived from radiation zones for normal operation (see DCD Chapter 12) assuming an expected 60 years of continuous operations with reactor power of 4,451 MW and steady state operating conditions (see DCD Chapter 1). The values for radiation during normal operation are specified in Table 5-1 as a function of in-plant location.

5.2 Abnormal Operating Conditions

Abnormal environments are defined to recognize possible plant service abnormalities that lead to short-term changes in environments at various equipment locations. Abnormal operating conditions are specified in Table 5-2 as a function of equipment location.

5.3 Seismic Events

See Attachment B

5.4 Containment Test Environment

Regulatory Guide 1.18 specifies that containment integrity is demonstrated at 1.15 times design pressure. The design pressure of the US-APWR containment is 68 psig. Consequently, the maximum pressure specified for the containment test is 59 x 1.15 = 78.2 psig. Other environmental parameters (such as temperature and humidity) of the containment test are adequately enveloped by the parameters specified for normal or abnormal plant conditions.

5.5 Design Basis Accident Conditions

Performance requirements are specified for those design basis accidents for which the equipment performs a safety-related function and which have a potential for changing the equipment environment due to increased temperature, pressure, humidity, radiation, or seismic effects. The environmental conditions for each applicable design basis accident are summarized in Table 5-3.

5.5.1 High-Energy Line Break Accidents Inside Containment

5.5.1.1 Radiation Environment – Loss-of-coolant Accident

The EQ radiation environment parameters of Total Integrated Dose and maximum dose rates will be determined for all compartments containing safety-related or important to safety equipment. Gamma and Beta sources will be addressed where applicable. Analyses of radiation environments will be performed in calculations. This is primarily a post accident evaluated radiation dose for inside the Containment. Radiation dose from containment is a maximum from the LOCA analysis results and impacts each piece of equipment differently. These impacts are determined by the individual piece of equipment's function, location, and safety class. This information is the result of engineering analysis for each piece of equipment that is safety-related or important to safety. Radiation sources can include both airborne activity in the containment and radioactivity containing equipment inside or outside of the containment. If necessary, particular equipment components may be subjected to more detailed evaluations based on their actual locations with respect to radiation sources.

Radiation doses associated with postulated accidents are determined by analytical computer codes as described in the DCD, Chapter 15.
The Nuclear source term for the LOCA accident analysis follows ANSI/ANS and NRC guidelines. Specifically, the guidance of 10 CFR 50.34 and NRC Regulatory Guide 1.183 are incorporated into the dose analysis. Beta radiation is also considered for component inside Containment (Zone 1).

The dose rate results (for each elevation inside the Containment and areas of the Auxiliary Building containing safety-related or important to safety equipment) are summarized in Dose Maps provided in DCD Chapter 12 at several times after the postulated accident (i.e., 1 hr, 1 day, 1 week, and 1 month). These show the gamma radiation levels in the areas from contained circulating post-accident fluids, and are intended to show that areas requiring post-accident accessibility are indeed accessible by operating personnel. Although they are not intended for EQ purposes, the radiological basis accident scenario used to develop these maps forms the basis to develop the time-integrated EQ gamma doses for up to 1 year of post-accident exposure, with sufficient time increments to allow consideration of particular equipment operational duration requirements, some of which are less than 1 year.

The values for radiation after the LOCA accident are specified in Table 5-4 as a function of in-plant location and time after accident. About in Zone 6 (Penetration Area and Safeguard Component Area (Radiological Area)), it is conservatively assumed that the radiation doses are equal to the values of in Zone 1.

5.5.1.2 Radiation Environment – Steam Line Break Accident

Sources associated with a steam line break accident are based on the release of reactor coolant system activity, assuming operation with the design basis fuel defect level of 1.0 percent. It is further assumed that an “event-initiated” iodine activity spike occurs, which increases the reactor coolant activity during the accident based on a rate of increase that is 500 times the normal activity appearance rate in the reactor coolant.

The activity inventory is instantaneously released into the containment (Zone 1) or the main steam piping area (Zone 10). It is conservatively assumed that the radiation doses in Zone 1 and Zone 10 resulting from a steam line break are equal to the values in Zone 1 for a loss-of-coolant accident.

5.5.1.3 Containment Pressure and Basis for Design

Maximum containment pressure transient is evaluated from the postulated LOCA analysis results and impacts each piece of equipment differently. The containment pressure is bounded by the large break LOCA discussed in the US-APWR DCD Chapter 6. Combined pressure curve considering various cases both for LOCA and MSLB in the containment general region is presented in Figure 5-1. It doesn’t include margin from IEEE 323-1974.

5.5.1.4 Containment Temperature and LOCA/MSLB Analysis

Containment temperature calculated from the postulated LOCA or MSLB analysis impacts each piece of equipment differently.
This temperature is defined as that seen by the equipment important to safety during an accident. The equipment important to safety is designed to function at the higher temperature for a time described in the DCD Chapter 6. The DCD contains tables and figures for specific temperature gradients to be used for EQ. Equipment located within the containment that is exposed to DBA would have its accident temperature determined by its location in the structure. Combined temperature curve considering various DBAs in the containment general region is presented in Figure 5-2. In this figure, margin required in IEEE 323-1974 is not included.

5.5.1.5 Indoor Chemical Environment – pH for Fluids

Indoor chemical environmental qualification requirements address exposure of SSCs to fluids inside the Containment Building post LOCA. These fluids can have an elevated pH that is damaging to SSCs. This requires the PEQO to perform engineering analysis and Equipment Qualification of SSCs to assure their performance maintains nuclear safety. The resulting EQ requirements for chemical exposure are provided in the procurement documents and specifications.

The concentration of chemicals used for qualification is equivalent to, or more severe than, that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling system initiation, or recirculation phase).

The PEQO is responsible for developing project level procedures to include these site-specific factors in the procurement effort.

5.5.1.6 Containment Flooding Analysis

Containment Flooding is from the LOCA analysis results and impacts each piece of equipment differently. These impacts are determined by the individual piece of equipment's function, location, and safety class. This information is the result of engineering analysis for each piece of equipment that is important to safety.

The flooding parameters, for purposes of qualification of equipment mounted within the standard plant, are defined as the potential flood levels inside the standard plant seismic category I buildings and structures as established by Section 3.4 of the US-APWR DCD Tier 2 Chapter 3.

5.5.2 High-Energy Line Break Accidents Outside Containment

Most of equipments outside containment are located in such regions that the normal operating environment maintained even if a high-energy line break occurs. On the other hand, several equipments important to safety are located in the main steam / feedwater piping area of the reactor building (RB) where harsh environment are attained by postulated SLB or FLB. The equipments that are qualified for the conditions resulting from such HELBs are required to operate and presented in Table 3D-2 in the DCD.

Figure 5-3 and 5-4 shows the combined design conditions for the equipments that are required to perform during and post SLB and FLB. Both of Figure 5-3 and 5-4 do not include margin from IEEE 323-1974.

The maximum pressure for any accident in the RB is 14.5 psig and maximum temperature is 327 $^{\circ}$F according to the results of GOTHIC calculation with multi-noding systems. Volumes of each compartment are underestimated by assuming loss volume is larger, compared to the actual design for conservatism. Parameters and options relevant to the junctions are set to provide large flow resistance between compartments. Passive heat sinks, blowout panels with the openings to the outer atmosphere and drain flow paths to the turbine building (TB) are considered as mitigation elements in the analysis. Condensation, natural convection and radiation heat transfer to the passive heat sinks are considered as mentioned in NUREG-0588 revision 1. 130 $^{\circ}$F of initial compartment vapor temperature is assumed in the analysis as a maximum value during operative or non-operative plant status presented in Table 5-1.

Followings are assumptions for released mass and energy which are considered as a boundary condition in the analysis.
Maximum 1ft^2 of the broken area is assumed for the both SLB and FLB analyses, which is described in SRP Branch Technical Position 3-3.
Released mass and energy regarding SLB is calculated by MARVEL-M which was incorporated to the containment analysis regarding MSLB in DCD Chapter 6.
For FLB, initial critical flow discharge based on the semi-steady state and subsequently discharged flow by the main feedwater pump are considered by simple assumption in order to calculate the released mass and energy.

Table 5-1 Normal Operating Environments (Sheet 1 of 3)

Location/Parameter	Normal Range	Notes
Zone 1 Containment		
Temperature	50 – 120 ºF	
Pressure	-0.2 – +1.0 psig	
Humidity	Non-condensing	
Radiation	1.0 rad/h	above the operation floor
Chemistry	None	
Zone 2 Main Control Room and Remote Shutdown Console Room		
Temperature	73 – 78 ºF	
Pressure	Atmospheric	
Humidity	25 – 60 %	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 3 Class 1E I&C Room		
Temperature	68 – 79 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 4 Class 1E Electrical Room, UPS Room, Battery Charger Room, and Reactor Trip Breaker Room		
Temperature	50 – 95 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 5 Class 1E Battery Room		
Temperature	65 – 77 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 6 Penetration Area and Safeguard Component Area (Radiological Area)		
Temperature	50 – 130 ºF	
Pressure	Slightly Negative	
Humidity	Non-condensing	
Radiation	100 rad/h	
Chemistry	None	

Table 5-1 Normal Operating Environments (Sheet 2 of 3)

Location/Parameter	Normal Range	Notes
Zone 7 Safety Related Component Area (Radiological Area)		
Temperature	50 – 130 ºF	
Pressure	Slightly Negative	
Humidity	Non-condensing	
Radiation	100 rad/h	
Chemistry	None	
Zone 8 Safety Related Component Area (Non-Radiological Area)		
Temperature	50 – 130 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 9 Essential Chiller Unit and Pump Room		
Temperature	50 – 105 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 10 Main Steam/Feedwater Piping Area		
Temperature	50 – 130 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 11 Gas Turbine Area		
Temperature	50 – 120 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	
Zone 12 Fuel Handling Area		
Temperature	50 – 105 ºF	
Pressure	Slightly Negative	
Humidity	Non-condensing	
Radiation	15 mrad/h	except for the inside of SFP
Chemistry	None	

Table 5-1 Normal Operating Environments (Sheet 3 of 3)

Location/Parameter	Normal Range	Notes
Zone 13 Auxiliary Building General Mechanical Area (Radiological Area)		
Temperature	50 – 105 ºF	
Pressure	Slightly Negative	
Humidity	Non-condensing	
Radiation	500 rad/h	Include the spent resin storage tank area
Chemistry	None	
Zone 14 Turbine Building General Mechanical Area		
Temperature	50 – 105 ºF	
Pressure	Atmospheric	
Humidity	Non-condensing	
Radiation	0.25 mrad/h	
Chemistry	None	

Table 5-2 Abnormal Room Conditions

Zone[1]/Room	Maximum Temperature	Humidity
Zone 1 Containment	150 ºF	Non-Condensing
Zone 2, 3, 4 and 5 All Rooms	122 ºF	Non-Condensing
Zone 8 EFW (T/D) Pump Room	175 ºF	Non-Condensing

Notes:
1. See Table 5-1 for environmental zone.

Table 5-3 Accident Environments

Zone[1]/Rooms	Parameter	
Zone 1	Temperature, Pressure	See Figure 5-x
Zone 2 Main Control Room	Pressure	Exceed + 0.125 inch w.g.[2]
	Humidity	Non-Condensing
Zone 6	Pressure	Below – 0.25 inch w.g.
Zone 10	Temperature, Pressure	See Figure 5-x

Notes:
1. See Table 5-1 for environmental zone.
2. Relative to all adjacent spaces to the control room envelope.

Table 5-4 Radiation Environments after LOCA Accident

Location	Accident Cumulative Dose (rad)			
	5minuites	2weeks	4months	1year
Zone 1 Containment	4.7E+05 (2.9E+06)	8.5E+07 (5.8E+08)	2.5E+08 (1.2E+09)	5.1E+08 (2.0E+09)
Zone 2 Main Control Room and Remote Shutdown Console Room	1.9E-01	2.8E+01	2.8E+01	2.8E+01
Zone 3 Class 1E I&C Room	1.9E-01	2.8E+01	2.8E+01	2.8E+01
Zone 4 Class 1E Electrical Room, UPS Room, Battery Charger Room, and Reactor Trip Breaker Room	1.9E-01	2.8E+01	2.8E+01	2.8E+01
Zone 5 Class 1E Battery Room	1.9E-04	3.6E-02	4.2E-02	5.1E-02
Zone 6 Penetration Area and Safeguard Component Area (Radiological Area)	4.7E+05 (2.9E+06)	8.5E+07 (5.8E+08)	2.5E+08 (1.2E+09)	5.1E+08 (2.0E+09)
Zone 7 Safety Related Component Area (Radiological Area)	9.2E+01	1.8E+04	2.1E+04	2.6E+04
Zone 8 Safety Related Component Area (Non-Radiological Area)	1.9E+01	3.6E+03	4.2E+03	5.1E+03
Zone 9 Essential Chiller Unit and Pump Room	1.9E-04	3.6E-02	4.2E-02	5.1E-02
Zone 10 Main Steam/Feedwater Piping Area	4.7E+05 (2.9E+06)	8.5E+07 (5.8E+08)	2.5E+08 (1.2E+09)	5.1E+08 (2.0E+09)
Zone 11 Gas Turbine Area	1.9E-04	3.6E-02	4.2E-02	5.1E-02
Zone 12 Fuel Handling Area	1.9E-01	2.8E+01	2.8E+01	2.8E+01
Zone 13 Auxiliary Building General Mechanical Area (Radiological Area)	1.1E+04	2.1E+07	1.8E+08	5.3E+08
Zone 14 Turbine Building General Mechanical Area	1.9E-04	3.6E-02	4.2E-02	5.1E-02

note) Cumulative dose in parentheses include Beta dose in zone 1, 6 and 10.

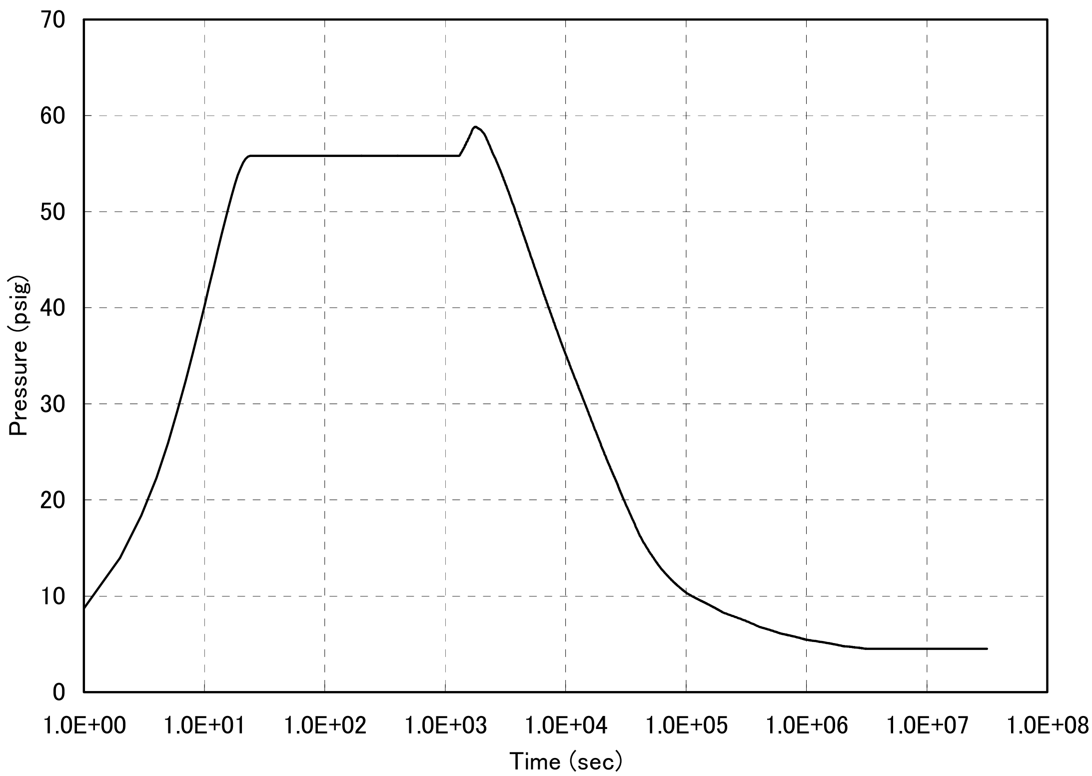


Figure 5-1 Environmental Curve for Containment Pressure

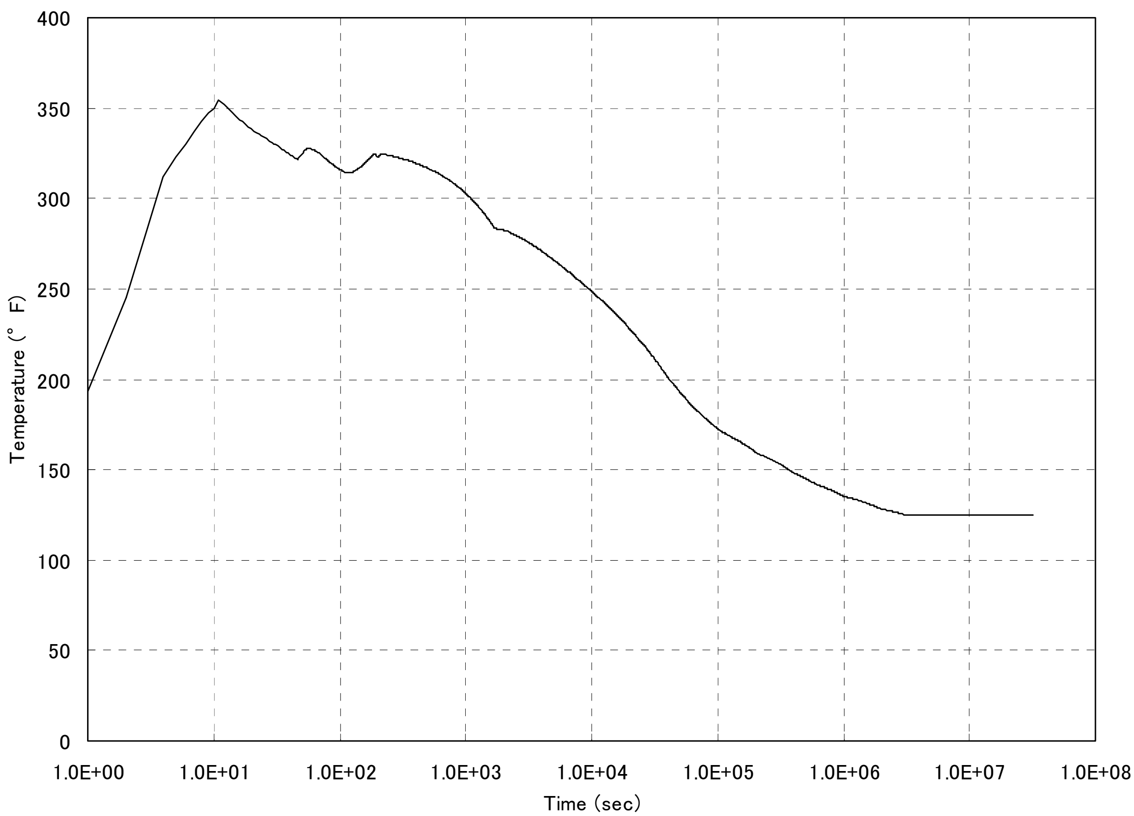


Figure 5-2 Environmental Curve for Containment Temperature

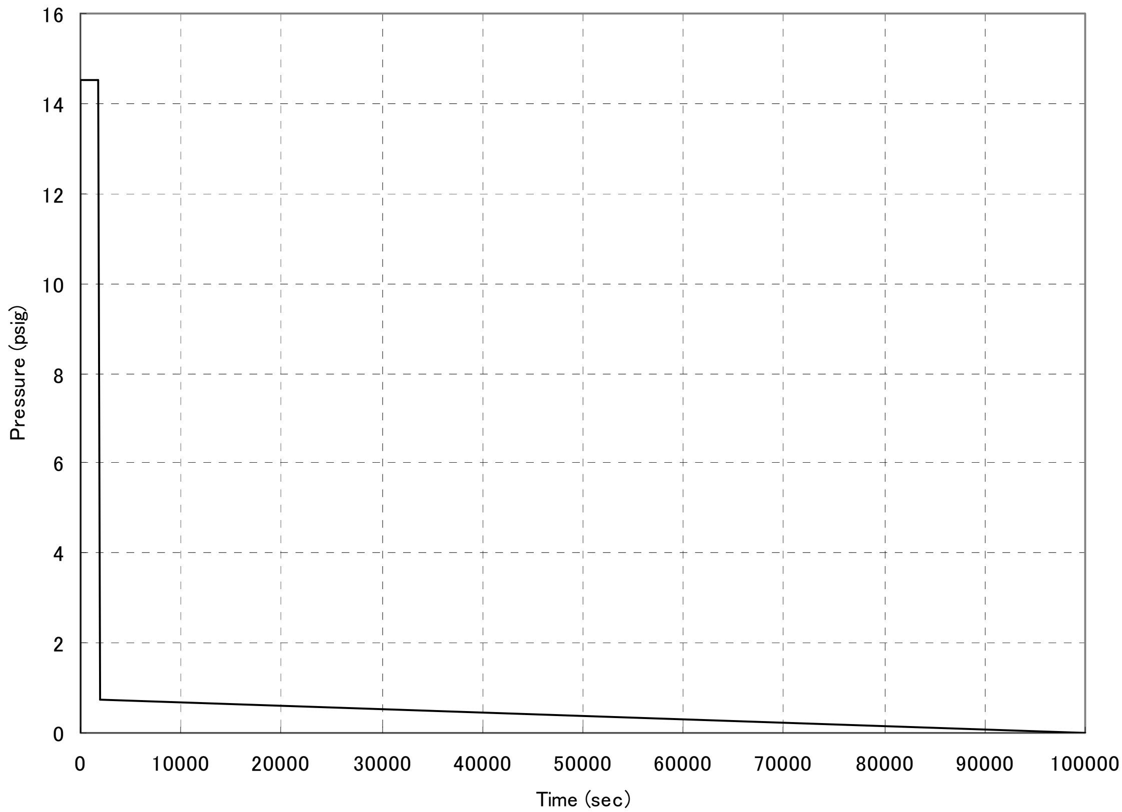


Figure 5-3 Environmental Curve for Pressure in MS/MF Piping Area

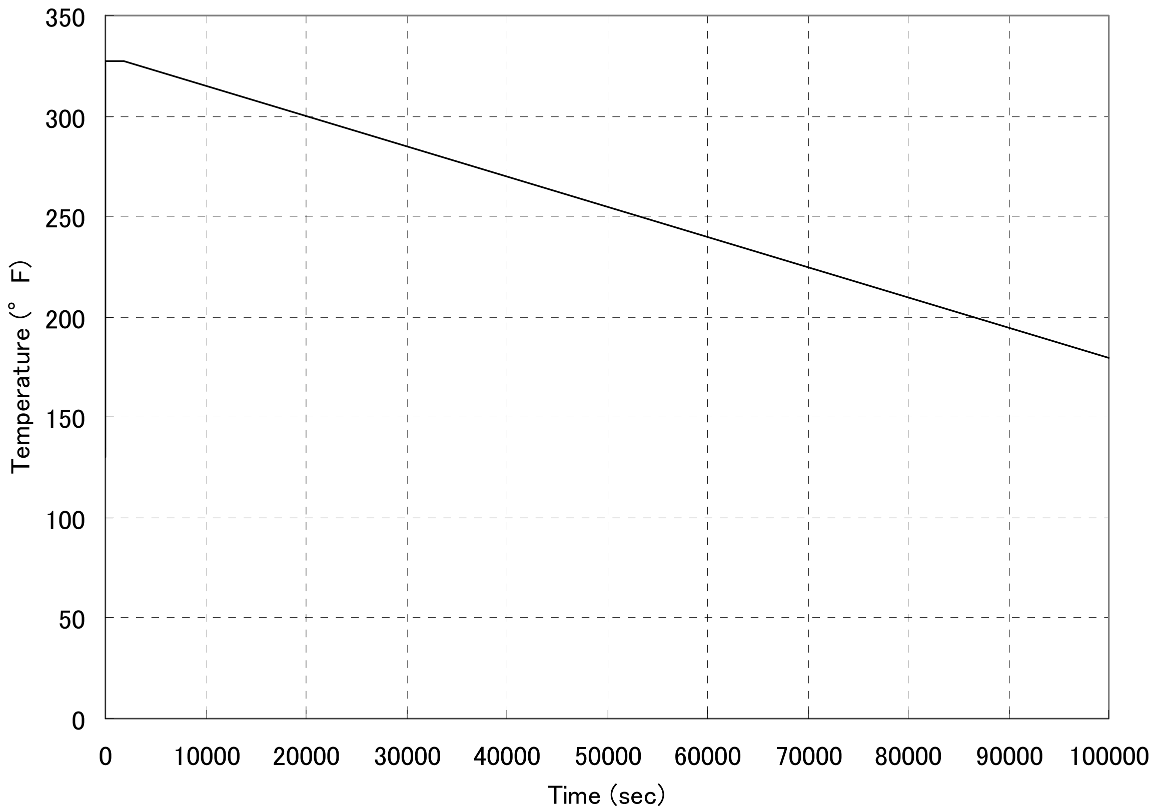


Figure 5-4 Environmental Curve for Temperature in MS/MF Piping Area

6.0 Qualification Methods

The recognized methods available for qualifying safety-related electrical equipment are established in IEEE 323. These are type testing, operating experience, analysis, on-going qualification, or a combination of these methods. The choice of qualification method for a particular item of equipment is based upon many factors. These factors include practicability, size and complexity of equipment, economics, and availability of previous qualification to earlier standards.

6.1 Type Test

The preferred method of environmental and seismic qualification of safety-related electrical and electromechanical equipment for the US-APWR equipment qualification program is type testing according to the guidelines and requirements of IEEE 323-1974 and 344-1987.

Additionally, qualification based on type tests performed according to IEEE 323 and 344, but not specifically for the US-APWR, may be used as a qualification basis. Section 6.5 discusses the combination of qualification methods as they apply to the US-APWR equipment qualification program (See subsection 6.5.1).

6.2 Analysis

Analysis can be used to demonstrate that equipment suffers no appreciable change in its ability to perform because of the environmental conditions associated with high stress events at any time in its qualified life. This method is generally limited to the following classes of equipment:

- Equipment that is simple in design and construction (e.g., cabinets, panels, instrument racks).

- Equipment where the DBA does not impose stresses additive to those imposed during normal operation in such a manner as to cause a common mode failure.

- Equipment that is similar to existing qualified equipment and where any differences are minor.

- Equipment that has no significant aging mechanisms over its qualified life.

6.2.1 Similarity

Similarity is employed to optimize equipment qualification. Representative samples of the model family being qualified are employed in the test sample. Supporting analysis is used to demonstrate that the results of the tests can be appropriately used to demonstrate the qualification of installed equipment. For example, the aging mechanisms of carbon resistors, printed circuits, junctions, solder joints, and wiring may not differ from one module to a similar module. If the qualified life of one module can be established, then modules of similar types will have an equivalent qualified life if modules have similar failure mechanisms. For the modules to be qualified, various types of equipment can be compared for similarity or grouping by comparing the following items:

- Type of technology used to design and manufacture the module.
- Type of critical components.
- Packaging, mounting, and type of connections.
- Service conditions.
- Safety functions.

For such a group, modules are type tested excluding aging. Some representative modules have an additional specimen type tested including aging. If the representative modules show no change in test results, whether aged or not aged, aging had no effect on safety function performance. Therefore, aging would have no effect on the safety function performance of the remainder of the similar group. However, if significant differences in performance between aged and unaged modules are found, similarity may not be used. In summary, the analysis to extrapolate qualified life for similar equipment includes the following:

- Group modules by similarity and justify the grouping.
- Type test modules, excluding aging.
- Type test one duplicate module from each similar group with aging.
- Determine if differences in results are acceptable for extending aging results to similar units.

6.2.2 Substitution

Substitution of parts or materials is acceptable if a comparison or analysis of their fit, form, and function supports the conclusion that the equipment performance is equal to or better than the originally qualified equipment.

6.2.3 Analysis of Safety-Related Mechanical Equipment

Environmental qualification of safety-related mechanical equipment is required to preclude common mode failures due to environmental effects of a design basis accident. Requirements are based on GDC 4 and 10 CFR 50, Appendix B. These criteria mandate that safety-related structures, systems, and components be designed to accommodate both normal and accident environmental effects.

6.3 Operating Experience

Qualification by experience is typically not employed in the US-APWR equipment qualification program as a prime method of qualification. Operating experience provides supportive evidence to the prime method of qualification. For those instances where seismic experience data are to be used, the Combined License applicant will provide documentation of the methodology. Where such information is provided, it is demonstrated that the experience is applicable to the safety-related functional requirements of the equipment. This demonstration of applicability includes an evaluation of operating environments, mountings, performance requirements, and performance history.

6.4 On-Going Qualification

The US-APWR equipment qualification program may employ on-going qualification through special maintenance and surveillance activities. However, this method of qualification is not suitable as a sole means for qualifying equipment for design basis accident conditions. On-going qualification, as a method, is used exclusively for safety-related equipment located in a mild environment area. Such use requires supplementary test, analysis, or experience data to address equipment operability and performance during and after a seismic design basis accident.

Documentation requirements for qualification that includes on-going qualification as a method are developed to conform with NRC guidance provided in Regulatory Guide 1.33, Revision 2.

6.5 Combinations of Methods

Qualification by a combination of the preceding methods is used whenever qualification by type test is not the sole basis of qualification under the US-APWR equipment qualification program. If analysis is used, justification includes identifying a test or experience bases, and addressing concerns related to departure from the required type test sequence.

6.5.1 Use of Existing Qualification Reports

Pre-existing qualification programs and documents are used only if the seismic test program satisfies the guidelines of IEEE 344-1987 and the environmental qualification program satisfies the guidelines of IEEE 323-1974.

Qualification test and analysis reports conforming to those IEEE, but not specifically performed to the US-APWR equipment qualification program parameters, may be acceptable as qualification bases. In such cases, supplementary qualification efforts described in subsections 6.2, 6.3, and 6.4 may be required to validate acceptability under the US-APWR equipment qualification program. Justifications are documented as analyses.

6.5.1.1 Aging

Past qualification tests may provide sufficient basis to preclude new aging simulation testing as part of the US-APWR program. Also, simulation of both electrical and mechanical operational cycling may be waived where existing data demonstrates equipment durability greatly in the excess of the estimated number of operating cycles for Class 1E service. Application of past qualification and other tests is considered in the development of test plans and analysis procedures. The bases and justification is provided in qualification documentation for cases where applicable aging parameters are omitted from the test sequence.

6.5.1.2 Seismic

Seismic qualification generally relies on analyses and justification to verify the adequacy or applicability of generic testing to a particular installed configuration of similar equipment. Analytical methods and documentation guidelines of IEEE 344-1987, as supplemented by Regulatory Guide 1.100, Revision 2, address these needs. Attachment B provides the US-APWR equipment qualification program requirements regarding seismic qualification.

6.5.1.3 High-Energy Line Break Conditions

Typically, existing qualification tests address conditions of high-energy line break environments occurring inside containment. These are used where it is demonstrated that the qualification envelops the applicable requirements.

7.0 REFERENCES

7.1 U.S. Regulations

10 CFR 50.34, “Contents of Applications, Technical Information,” Subsection (f) (2) (ix).

10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

10 CFR 50.67, “Accident Source Term.”

10 CFR 50, Appendix A, General Design Criterion 1, “Quality Standards and Records;” General Design Criterion 2, “Design Basis for Protection against Natural Phenomena;” General Design Criterion 4, “Environmental and Dynamic Effects Design Basis;” and General Design Criterion 23, “Protection System Failure Modes.”

10 CFR 50, Appendix B, Criterion III, “Design Control;” Criterion XI, “Test Control;” and Criterion XVII, “Quality Assurance Records.”

10 CFR 52.47, “Contents for Applications.”

10 CFR 52.97, “Issuance of Combined Licenses.”

7.2 U.S. Regulatory Guides

NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.”

NUREG/CR-6842, “Advanced Reactor Licensing: Experience with Digital I&C Technology in Evolutionary Plants.”

NUREG/CR6901, “Guidelines for Electromagnetic Interference Testing in Power Plants.”

NUREG-0737, “Clarification of TMI Action Plan Requirements.”

NUREG-800 SRP 3.10, “Seismic and Dynamic Qualification of Mechanical & Electrical Equipment.”

NUREG-800 SRP 3.11, “Environmental Qualification of Mechanical and Electrical Equipment.”

NUREG-800 SRP Branch Technical Position 3-3, “Protection against Postulated Piping Failure in Fluid Systems Outside Containment”

Regulatory Guide 1.30, (Safety Guide 30), “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.”

Regulatory Guide 1.40, “Qualification Tests for Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.63, “Electric Penetration Assemblies in Containment Structures for Light Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.73, “Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants.”

Regulatory Guide 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.”

Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

Regulatory Guide 1.131, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants.”

Regulatory Guide 1.151, “Instrument Sensing Lines.”

Regulatory Guide 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants.”

Regulatory Guide 1.158, “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants.”

Regulatory Guide 1.180, “Guidelines for Evaluation Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.”

Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

Regulatory Guide 1.139 “Guidance for Residual Heat Removal” (for Comment Rev. 0, May 1978) Note: Cold shutdown requirements as related to environmental qualification of equipment Conformance with exceptions. Criterion 7 applies to a site-specific operational program.

Regulatory Guide 1.156 “Environmental Qualification of Connection Assemblies for Nuclear Power Plants” (Rev. 0, November 1987) Conformance with no exceptions identified.

Regulatory Guide 1.158 “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants” (Rev. 0, February 1989) Conformance with no exceptions identified.

Regulatory Guide 1.189, “Fire Protection for Nuclear Power Plants.”

Regulatory Guide 1.209, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants.”

DG-1175, “Seismic Qualification of Electric and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants (ML072620346)”, 5/2008

7.3 Regulatory Review Precedent

Letter dated December 17, 1996, from L.E. Martin, Houston Lighting & Power, to the U.S. Nuclear Regulatory Commission, “South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, 10 CFR 50.59 Summary Report.”

Letter dated April 8, 1998, from Thomas Alexion, NRC, to William Cottle, STP Nuclear Operating Company. “Request for Additional Information on Elimination of Environmental Qualification of Mechanical Components. South Texas Project, Units 1 and 2 (STP) (TAC Nos. M98912 and M98913).”

Letter dated May 6, 1998, from S.E. Thomas, STP Nuclear Operating Company, to U.S. Nuclear Regulatory Commission, “Response to Request for Additional Information on Elimination of Environmental Qualification of Mechanical Components,” Docket Nos. STN 50-498, STN 50-499, Units 1 and 2 (STP).

Letter dated September 24, 1998, from Thomas Alexion, NRC to PD IV-1 File, “Licensee’s 10 CFR 50.59 Evaluation of Elimination of Environmental Qualification of Mechanical Components, South Texas Project, Units 1 and 2 (STP) (TAC Nos. M98912 and M98913).”

7.4 U.S. Industry Codes and Standards

IEEE Std C37.82-2004, ‘IEEE Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Power Generating Stations.”

IEEE Std C37.105-1987, “IEEE Standard for Qualifying Class 1E Protective Relays and Auxiliaries for Nuclear Power Generating Stations.”

IEEE Std 7.4.3.2-2003, “IEEE Standard Criteria for Digital Computers in Safety System of Nuclear Power Generating Stations.”

IEEE Std 317-1983 (reaffirmed 1992),” IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations.”

IEEE Std 323-1974,” IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.”

IEEE Std 334-1971, “IEEE Trail-Use Guide for Type Tests for Continuous-Duty Class 1 Motors Installed Inside The Containment of Nuclear Power Generating Stations.”

IEEE Std 344-1975, “IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.”

IEEE Std 381-1977 (reaffirmed 1984), “IEEE Standard Criteria for Type Tests of Class 1E Modules Used in Nuclear Power Generating Stations.”

IEEE Std 382-1972, “IEEE Trial-Use Guide for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Stations.”

IEEE Std 383-2003, “IEEE Standard for Type Test of Class 1 Electric Cables and Field Splices for Nuclear Power Generating Stations.”

IEEE Std 387-1995, “IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.”

IEEE Std 497-2002, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations.”

IEEE Std 535-1986, “IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations.”

IEEE Std 572-1985, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations.”

IEEE Std 627-1980 (reaffirmed 1991), “IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations.”

IEEE Std 628-2001, “IEEE Standard Criteria for Design, Installation and Qualification of Raceway Systems.”

IEEE Std 638-2006, “IEEE Standard for Qualification of Class 1E Transformation Nuclear Power Generating Station.”

IEEE Std 649-1980, “IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations.”

IEEE Std 649-2005, “IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations.”

IEEE Std 650-2006, “IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations.”

IEEE Std 1202-1991, “IEEE Standard for Flame Propagation Testing of Wire and Cable.”

IEEE Std 1205-2000, “Guide for Assessing, Monitoring, and Mitigating Aging Effects on Class 1E Equipment used in Nuclear Power Generating Stations.”

IEEE Std 1290-1996, “IEEE Guide for Motor Operated Valve (MOV) Motor Application, Protection, Control, and Testing in Nuclear Power Generation Stations.”

ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications”, 1994.

ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants”

7.5 Industry Group References

EPRI, NP-5652, “*Guidelines for the Utilization of Commercial Grade Items in Nuclear Safety Related Applications (NCIG-07),” 1988*

EPRI, TR-102260, "*Supplemental Guidance for the Application of EPRI Report Np-5652 on the Utilization of Commercial Grade Items,*" 1994

EPRI, TR-106439, "*Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications,*" 1996

EPRI, TR-1001452, "Generic Qualification of Commercial Grade Digital Devices" Lessons Learned, 2001.

EPRI, TR 1001468, "Generic Qualification of the Rosemount 3051N Pressure Transmitter, Summary of Activities and Results," 2001

EPRI, TR-112579, "Critical Characteristics for Acceptability of Seismic Sensitive Items (CCASSI), 2000

EPRI , TR-1003105, "Dedicated Commercial-Grade Items Procured from ISO 9000 Suppliers," 2001

EPRI-TR-102323, Revision 3, "Guidelines for Electromagnetic Interference Testing for Power Plants," 2004 (1003697)

EPRI-TR-100516, "Nuclear Power Plant Equipment Qualification Reference Manual," 1992

INPO EPG-02, 2005 Engineering Program Guide, "Environmental Qualification of Electrical Equipment." Note: This document primarily directed to operating utilities dealing with long term EQ programs, however, it provides very good insight into formulization of the PEQO.

NUPIC (Nuclear Procurement Issues Committee) Joint Commercial Grade Survey Program Description, NUPIC Document No. 6 Joint Audit Program 2001 (see http://www.nupic.com/)

Nuclear Industry Assessment Committee (NIAC)) Audit programs and assessments of nuclear suppliers using member programs, NIAC Checklists, govern assessments and audit (NIAC Audit Checklist, Rev. 6)

7.6 MHI Documents

MNES US-APWR QAPD SQ-QD-070001 US-APWR Quality Assurance Program Description

US-APWR Topical Report, "Quality Assurance Program Description," DQD-HD-19005, Rev. 0 (MUAP-07002 and -07003 Rev.0), January 2007

Attachment A Summary of Statutes, Regulatory Guides, Industry Codes and Standards Applicable to the US-APWR EQP

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
Federal Statutes	10 CFR 50 Energy	Defines Legal Requirements for Nuclear Plant Licensure	Invoked in sections for specific licenses
10 CFR 50, App A GDC 1	Quality Standards and Records	Basis for Nuclear Plant QA Program	Nuclear Plant QA is the basis for the EQ Program
10 CFR 50, App A GDC 2	Design Basis for Protection Against Natural Phenomena	Design Basis for engineering safety analysis	Design Basis for engineering sets the environmental and performance requirements for the EQ Program
10 CFR 50, App A GDC 4	Environmental and Dynamic Effects Design Basis	Expansion of requirements for GDC 2 Natural Phenomena	Sets seismic and other design basis requirements for EQ Program
10 CFR 50, App A GDC 17	Electric Power Systems	Provides interface requirements for Nuclear Plant electric power systems	Sets engineering and performance design basis for electric power systems
10 CFR 50, App A GDC 19	Control Room	Define control room requirements including operating during DBA	Requires equipment be qualified to operate in DBA environments
10 CFR 50, App A GDC 20	Protection Systems Functions	Sets specific system functions for NSSS control systems	This is a major control system for the new nuclear plant with EQ Program performance requirements
10 CFR 50, App A GDC 21	Protection System Reliability and Testability	Sets test requirements for NSSS control systems	This sets EQ Program test and turnover requirements
10 CFR 50, App A GDC 22	Protection System Independence.	Prevents control and protection functions from interfering with protective function	Requires EMI/EMF evaluations to assure no interference
10 CFR 50, App A GDC 23	Protection System Failure Modes.	Sets criteria for control room including certain operation during accidents	Requires elements of control room to withstand some radiation exposure
10 CFR 50, App A GDC 24	Separation of Protection and Control Systems	Sets separation engineering and design requirements for protection and control systems	This GDC 24 has little input to EQ Program; it is included only for reference to multiple train requirements for same equipment subject to EQ Program

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
10 CFR 50, App A GDC 29	Protection Against Anticipated Operational Occurrences	Sets engineering and design requirements for protection and control system power operation	This GDC 29 has little input to EQ Program; it is included only for reference to power operation
10 CFR 50, App A GDC 30	Quality of Reactor Coolant Pressure Boundary	High quality Reactor Coolant System (RCS) components and construction	Verify materials are qualified for environment used in reactor coolant pressure boundary
10 CFR 50.49	Qualification of Electrical Equipment Important to Safety	Assure equipment will function to mitigate a DBA	Mechanical equipment implied
Regulatory Guides		Provide guidance in meeting regulatory requirements	Invoked in sections for specific licenses
Regulatory Guide 1.22	Periodic Testing of Protection System Actuation Function	Set specific system functions for NSSS control systems	This is a major control system for the EQ Program performance requirements
Regulatory Guide 1.29	Seismic Design Classification	Provides guidelines for seismic design classification so that those SSCs which need to withstand an SSE can be identified	Seismic design is an EQ Program critical characteristic of safety-related or important to safety SSC
Regulatory Guide 1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	Sets testing criteria for EQ and acceptance for motors inside Containment for LOCA analysis	EQ Program critical characteristic of safety-related or important to safety electrical components
Regulatory Guide 1.63	Electric Penetration Assemblies in Containment Structures for Light Water-Cooled Nuclear Power Plants	IEEE Std 317-1983 endorsed by NRC	This standard prescribes requirements for the design, construction, testing, qualification, and installation of electric penetration assemblies in containment structures for stationary nuclear power generating stations
Regulatory Guide 1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	Initial Test Programs	SSCs to be tested to satisfy the requirements of GDC-1, “Quality Standards and Records”

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
Regulatory Guide 1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	Sets acceptance values for electric valve operators installed inside the Containment	EQ Program captures the critical requirements for electric valve operators installed inside the Containment
Regulatory Guide 1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	Sets seismic and other dynamic qualification of safety-related or important to safety equipment	Three categories of electric equipment important to safety are qualified for their application and specified performance and provide requirements for establishing environmental qualification methods and qualification parameters. These are (1) safety-related electric equipment (Class IE), (2) non-safety-related electric equipment (non-Class IE) whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by safety-related equipment, and (3) certain post accident monitoring equipment
Regulatory Guide 1.148	Function Specification for Active Value Assemblies in Systems Important to Safety in Nuclear Power Plants	Sets requirements for a functional specification	Ensures operability of active value assemblies
Regulatory Guide 1.97	Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident		This endorses IEEE Std 497-2002
Regulatory Guide 1.100 & DG 1175	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants	Safety-related equipment covered under the EQ Program added seismic requirements	DG-1175 endorses, with exceptions and clarifications, IEEE Std 344 – 2004 and, for the first time, ASME QME-1.

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
Regulatory Guide 1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	Safety-related equipment covered under the EQ Program	Refer to the DCD Section 8.1.5.3.
Regulatory Guide 1.209 & DG-1077	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	Safety-related or important to safety electrical equipment, which consists of computer-based I&C systems, located in a mild environment is covered	This endorses IEEE Std 323-2003
Regulatory Guide 1.183 & DG 1081	Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	This Regulatory Guide provides the basis for the radiation source term for evaluating design basis accidents	EQ Program radiation doses input for safety-related or important to safety equipment analysis
Regulatory Guide 1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	Qualification of battery chargers	IEEE Std 650-2006, “IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations,” endorsed by this regulatory guide with some exceptions
NRC Clarification Letters			
Industry Standards			
ASME Section III	American Society of Mechanical Engineers Boiler and Pressure Vessel Code	Addresses mechanical components and systems important to safety and the qualification needed to meet the mechanical requirements of that code	Seismic analysis criteria and methods for the mechanical and piping systems for the US-APWR. Note that active components are under ASME QME-1, very little Section III piping is within the EQP scope, however, certain elements such as anchor bolts fall within the EQP scope boundary
ASME NQA-1 1994	American Society of Mechanical Engineers Quality Assurance Requirements for Nuclear Facilities	Quality Assurance Program Requirements	Augments EQP record keeping generation and maintenance and other QA functions.

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
ASME QME-1-2007	American Society of Mechanical Engineers Qualification of Active Mechanical Equipment Used in Nuclear Power Plants	Provides the requirements and guidelines for qualification of active mechanical equipment	Scope is for mechanical equipment whose function is required to ensure the safe operation of safe shutdown of a nuclear power plant
IEEE Std 344-1987, 2004	Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	Industry standard	Endorsed by NRC in Regulatory Guide 1.89
IEEE Std 323-1974, 2003	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations	Industry standard	Endorsed by NRC in Regulatory Guide 1.89 (1.100)
IEEE Std 535 1986	IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations	Industry standard	Endorsed by NRC in Regulatory Guide 1.58
NUREG Series Publications			
NUREG 0800	Standard Review Plan 3.10, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	NUREG 0800 addresses the preparation of the US-APWR DCD and COLA FSAR and the licensing requirements therein	Sets forth details on EQ Program requirements for licensing by U.S. NRC
NUREG 0800	Standard Review Plan 3.11, Environmental Qualification of Mechanical and Electrical Equipment	NUREG 0800 addresses the preparation of the US-APWR DCD and COLA FSAR and the licensing requirements therein	Sets forth details on EQ Program requirements for licensing by U.S. NRC
Industry Practices			
EPRI reports	TR 102260	EPRI and NRC acceptance of commercial grade dedication pursuant to NRC guidelines	Filter to see if commercial grade items are acceptable

DOCUMENT NUMBER	TITLE	PURPOSE	COMMENTS
EPRI reports	TR-106439 Guideline on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Application	Prepared by operating utilities to address procurement of commercial grade items	Process accepted by NRC as delineated in TR-102260
	TR-1003585 Generic Qualification/Dedication of Digital Components	Basis for using commercial grade digital instrument and controls	IEEE supporting documents
	TR-1003105 Dedicating Commercial-Grade Items Procured From ISO 9000 Suppliers	New process for using ISO 9000 suppliers	
	TR-017218 Guideline for Sampling in the Commercial grade Item Acceptance Process	Example of commercial grade dedication procedures	
	TR-112579 Critical Characteristics for Acceptance of Seismically Sensitive Items		
	NUPIC Audit Procedures	Provides audit of nuclear suppliers to show general compliance for 10 CFR 50, APP B QA requirements	See NAIC, utility initiative, NUPIC uses NUPIC audits and shares results. NUPIC methodologies acceptable to NRC.
	NAIC Audit Procedures	Provides audit of nuclear suppliers to show general compliance for 10 CFR 50, APP B QA requirements	See NUPIC, NAIC uses member audits and shares results. Process acceptable to NRC on a utility by utility basis.

Attachment B

Equipment Seismic Qualification Program Description

B.1 General Description

The program for seismic and dynamic qualification of equipment consists of procedures and criteria which are governed by and form a part of the overall MHI US-APWR Equipment Qualification (EQ) Program.

The overall EQ Program is comprised of the US-APWR Equipment Qualification Program Technical Report and EQ program directives and procedures that define the programmatic requirements for all environmental aspects of equipment qualification, including seismic / dynamic, radiation, pressure, temperature, humidity, aging, flooding, chemical, and synergistic effects. With respect to seismic/dynamic qualification, the program directives and procedures address topics such as:

- Program management structure
- Documentation and records retention and management
- Methods of qualification
- Quality assurance
- Personnel qualifications and training
- Implementation of the program during various phases (design, procurement, construction, startup/commissioning, initial operations, and turnover
- Preparation, maintenance, and control of equipment qualification files

The EQ Program procedures also contain, or give direction on, sample document formats which can be used in implementing the equipment seismic qualification program. These include:

- Equipment Seismic Qualification Reports (EQSR)
- Equipment Qualification Summary Data Sheets (EQSDS), which can be used for entry of seismic data into a project equipment qualification database
- Checklists for review of vendor/supplier seismic qualification reports

A complete listing and description of EQ Program directives, procedures, and their attachments addressing the above topics is given in Section 6.0 and Attachment A of this Technical Report.

B.2 Scope of Equipment Seismic Qualification and Seismic Qualification Criteria

Seismic category I equipment is required to be seismically and dynamically qualified under the program by demonstrating that its structural integrity is maintained and that it is capable of performing its designated safety function during and after a postulated earthquake in conjunction with the full range of applicable normal and accident loads and conditions.

Seismic category I equipment requiring qualification in accordance with the US-APWR EQ Program includes:

- Equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment reactor heat removal.
- Equipment essential to preventing significant release of radioactive material to the environment.
- Instrumentation (including accident and post-accident monitoring) needed to assess plant and environs conditions during and after an accident, as described in USNRC Regulatory Guide (RG) 1.97 “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

The equipment seismic qualification program criteria define specific technical requirements for seismic and dynamic qualification of seismic category I, safety-related mechanical equipment (excluding piping), and seismic category I (class 1E) electrical and instrumentation equipment, including associated supports and mountings. The program includes qualification of Category I tanks and reservoirs for hydrodynamic seismic loads, where applicable. All such equipment that is required to perform functionally or maintain its structural integrity, as described above, is subject to rigorous seismic/dynamic qualification. A detailed listing of MHI US-APWR standard plant seismic category I equipment, requiring seismic qualification, is given in Table 3.2-2 and Appendix 3D in Tier 2 of the MHI US-APWR DCD.

It should be noted that detailed criteria for functionality testing and inspection of mechanical and electrical equipment such as performance tests, hydrostatic tests, leakage tests, etc. are not within the scope of the equipment seismic qualification program. Also, qualification through dedication of commercial grade items (i.e., those items which are available commercially and not designed and manufactured under a quality assurance program complying with 10 CFR 50 Appendix B) is not within the scope of the equipment seismic qualification program or the overall generic EQ Program. For commercial grade items that will be used in safety-related and important to safety applications, a commercial grade dedication plan and special technical evaluations are required which account for the critical design and acceptance characteristics of the items. MHI/MNES may utilize commercial grade dedication as appropriate for the project specific EQ Program.

B.3 Important to Safety and Seismic Category II Qualification Requirements

The equipment seismic qualification program criteria also define technical requirements for seismic and dynamic qualification of important to safety equipment whose failure could prevent satisfactory accomplishment of one or more of the safety-related functions listed in Section 4.10.2 above.

This includes seismic category II equipment, defined as that equipment which performs no safety-related function, and whose continued function is not required, but whose structural or functional failure or interaction could degrade the functioning or integrity of a seismic category I SSC to an unacceptable level, or could result in incapacitating injury to occupants of the control room.

Therefore, seismic category II equipment can be seismically qualified by demonstrating that it retains its position sufficiently in the event of an SSE to the extent that it will not cause unacceptable structural interaction with or failure of seismic category I SSCs. For fluid systems, this requires an appropriate level of pressure boundary integrity to prevent seismically-induced flooding that may cause adverse effects on safety-related SSCs.

Note that in cases where it is not possible or practical to isolate the seismic category I equipment, non-seismic equipment that is adjacent to seismic category I equipment is classified as seismic category II and analyzed and supported such that an SSE event does not cause an unacceptable interaction with the seismic category I equipment.

Based on the qualification objectives defined above for seismic category II equipment and supports, the degree of seismic qualification for seismic category II SSCs does not warrant the full extent of sophisticated dynamic analysis or seismic vibration testing that are typically applied for qualification of seismic category I equipment. Simplified analytical techniques such as the equivalent static method are acceptable for demonstrating structural integrity of seismic category II equipment and supports. However, more sophisticated dynamic analyses can be applied in some cases where demonstration of pressure boundary integrity is required. Analysis of seismic category II equipment and supports under the scope of the US-APWR EQ program shall conform to the requirements for seismic analysis established in Chapter 3 Section 3.7 of the US-APWR DCD.

Criteria for qualification of seismic category II equipment are presented in MHI US-APWR criteria entitled "Seismic Qualification of Category II Electrical and Mechanical Equipment (including supports)." EQ Program Procedure 6, "US-APWR EQ Program Analysis Requirements" also includes seismic analysis requirements as part of its detailed program.

B.4 Codes and Standards for Seismic / Dynamic Qualification

The program for seismic and dynamic qualification complements, and is consistent with, the technical requirements and parameters that are specific of Tier 2 of the MHI US-APWR Design Control Document, particularly those of Chapter 3 Sections 3.7, 3.10 and 3.11. The equipment seismic qualification program technical requirements are based largely on those contained within IEEE Std 344 and ASME QME-1 (for functional qualification of active mechanical equipment). Requirements of IEEE Std 323 that are pertinent to seismic and dynamic qualification are incorporated into the program for equipment seismic qualification.

The equipment seismic qualification program also incorporates supplementary seismic and dynamic qualification requirements from United States Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.100, Draft Regulatory Guide DG-1175, RG 1.148, NUREG-0800 Standard Review Plan 3.10, NUREG-0800 Standard Review Plan 3.11, USNRC "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications."

The equipment seismic qualification program conforms to the requirements of Appendix S "Earthquake Engineering Criteria for Nuclear Power Plants" of Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50) and applicable portions of the general requirements contained in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

Sections of 10 CFR Part 50 Appendix A "General Design Criteria for Nuclear Power Plants" also provides general requirements for equipment seismic qualification including, but not limited to, the following General Design Criteria:

- General Design Criterion (GDC) 1, “Quality Standards and Records”
- GDC 2, “Design Bases for Protection Against Natural Phenomena”
- GDC 4, “Environmental and Dynamic Effects Design Basis”
- GDC 14, “Reactor Coolant Pressure Boundary”
- GDC 22, “Protection System Independence”
- GDC 30, “Quality of Reactor Coolant Pressure Boundary”

Seismic category I and II SSCs must also meet the pertinent QA requirements of 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” related to all activities associated with design, procurement, fabrication, construction, inspection, and/or testing, including but not limited to:

- Criterion III, “Design Control”
- Criterion XI, “Test Control”
- Criterion XVII, “Quality Assurance Records”

The equipment seismic qualification program and the overall MHI US-APWR EQ program conform to the applicable quality assurance requirements of ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications.”

The above-cited codes, standards, and regulatory documents are not intended to be an all-inclusive list. Refer to Attachment B of this Technical Report for a complete list of references applicable to the equipment seismic qualification program and the overall MHI US-APWR EQ Program.

B.5 Required Response Spectra Used for Seismic Qualification

The required response spectra (RRS) that in IEEE Std 344 are defined as the spectra which represent the motion at the support of the seismic category I and II equipment serve as the basic seismic input/parameters for equipment to be qualified under the auspices of the US-APWR Equipment Qualification Program. The broadened SSE and Operating Basis Earthquake (OBE) in-structure response spectra (ISRS) corresponding to the location(s) and elevation(s) of a particular piece of equipment within a seismic category I building or structure define the seismic input motion to be used for qualification of that equipment. For equipment that is supported on foundations resting on the ground surface, the corresponding Foundation Input Response Spectra (FIRS) shall serve as RRS for their seismic qualification. Multiple sets of ISRS may be applicable to a single piece of equipment because it may be attached at several elevations, or because it may be part of a system that extends across multiple building/structure elevations and locations. In these cases, equipment seismic qualification considers the effects of differential support motions. Multiple sets of ISRS may be applicable to a particular model or type of equipment because it may be used within multiple buildings or at multiple locations within a building or structure.

As per EQ program procedural requirements, MHI/MNES must be contacted for resolution if at any time it becomes apparent that equipment to be qualified will be mounted at a location or elevation for which there are no corresponding ISRS. This condition can occur during detailed design or during the procurement phase as particular equipment mounting characteristics or positions/locations are determined. This can also occur during the construction phase of a project when field conditions require a certain SSC to be relocated. The PEQP procedures address these requirements.

The RRS may be subsequently converted to time histories for use as input for numerical time-history analyses or for setting shake table input motion. In so doing, the direction of US-APWR equipment seismic qualification program is that the time histories should be generated in accordance with the general guidance of NUREG 0800 SRP 3.7.1, with a Nyquist frequency of 100 Hz.

B.6 Standard Plant versus Site-Specific SSE RRS

The SSE used for seismic qualification of equipment is defined as that design earthquake which produces the maximum vibratory ground motion for which structures, systems and components, which perform a primary safety function, are designed to remain functional. The equipment seismic qualification for the US-APWR EQ program uses two different types of SSE:

1) A site-independent SSE used for the qualification of US-APWR standard plant equipment where ground design motion is represented by the certified seismic design response spectra (CSDRS); and
2) A site-dependent (site-specific) SSE used for qualification of non-standard portions of the US-APWR where ground design motion is represented by site-dependent ground motion response spectra (GMRS) and Foundation Input Response Spectra (FIRS).

For standard plant seismic category I buildings and structures, the design does not vary and therefore the ISRS used as input for seismic qualification will be the same for each building and structure at every US-APWR site. US-APWR standard plant seismic category I buildings and structures include, but are not limited to, the RB, pre-stressed concrete containment vessel (PCCV), containment internal structure, and east/west power source buildings (PS/BS). For these buildings and structures, the applicable SSE ISRS are the SSE ISRS presented in "Dynamic Analysis of the Coupled RCL-RB-PCCV-CIS Lumped Mass Stick Model, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008" and "Seismic Design and Analysis of Power Source Buildings, MUAP-08002, Mitsubishi Heavy Industries, Ltd., February 2008."

For equipment located in seismic category I buildings and structures that are not part of the US-APWR standard plant and are designed on a site-specific basis, SSE ISRS and FIRS are established on a site-specific basis. The ground motion response spectrum associated with the site-specific SSE must be enveloped by, but may have a different shape than, the standard plant CSDRS. As required by the US-APWR DCD Tier 2 Chapter 3, the site-specific SSE ISRS are to be developed based on the site-specific SSE response analysis using seismic analysis methods that are consistent with Section 3.7 of the DCD and with particular requirements for ISRS given in DCD. For purposes of equipment qualification, standard plant SSE ISRS may be used for qualification of equipment located in site-specific buildings and structures, provided that it can be demonstrated that the applicable site-specific SSE ISRS are fully enveloped (both shape and magnitude) by the SSE ISRS used for the qualification, for all pertinent frequencies. In this case, the qualification documentation must identify this approach and clearly demonstrate how the applicable site-specific SSE ISRS are enveloped by the standard plant SSE ISRS. This approach may also be used for considering OBE ISRS – see Section 4.10.7, below.

As per the US-APWR equipment seismic qualification EQ programmatic requirements, it is the responsibility of the PEQO to ensure that appropriate ISRS are identified and included in the equipment vendors/suppliers procurement specifications for their use in seismic qualification of their equipment.

B.7 Consideration of OBE and Application of OBE ISRS for Equipment Seismic Qualification

The operational basis earthquake (OBE) specifies the magnitude of ground motion that requires plant shutdown. Consistent with Tier 2 Chapter 3 Section 3.7 of the MHI US-APWR DCD, the OBE is set on a site-specific basis by the COL applicant for each individual US-APWR project, but must be enveloped by 1/3 of the standard plant CSDRS.

Appendix S of 10 CFR 50 stipulates that the magnitude of an OBE can be adopted either as (A) 1/3 or less of the SSE; or (B) a value greater than 1/3 of the SSE. For Option A, explicit response or design analyses considering the OBE are not required to be performed for plant SSCs. If Option B is chosen, explicit analysis and design must be performed to demonstrate that all SSCs necessary for continued operation without undue risk to the health and safety of the public will remain functional within applicable stress, strain, and deformation limits.

Subsection 3.7.1 of the MHI US-APWR DCD has set OBE for design of the US-APWR standard plant at 1/3 of the SSE. Therefore, for purposes of standard plant equipment qualification, explicit analysis of standard plant equipment for OBE is not required.

However, it is recognized that during the life of the plant, equipment may be subjected to seismic excitations at lower levels than the SSE, which has the potential to reduce the “life expectancy” of those items sensitive to fatigue. Therefore, to account for fatigue effects, analysis and testing shall include the equivalent effects of five one-half SSE events (based on the standard plant SSE or site-specific SSE, as applicable) followed by one full SSE event (10 full cycles of the maximum SSE stress range). Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycle for five one-half SSE events may be used in accordance with Annex D, “Test Duration and Number of Cycles,” to IEEE Std 344-2004, when followed by one full SSE. This is consistent with guidance given in USNRC SECY-93-087 and DG-1175. When considering OBE ISRS for purposes of fatigue during seismic qualification testing of US-APWR standard plant equipment, it is acceptable to obtain OBE spectra by scaling directly from the site-specific SSE ISRS.

For seismic qualification of equipment located in site-specific parts of US-APWR plant, the site-specific OBE RRS must be considered. Similar to the approach for the standard plant, if OBE is set at 1/3 or less of the site-specific SSE, then for purposes of equipment qualification, explicit design and analysis of the site-specific equipment for OBE is not required (except for fatigue considerations). If, however, the site-specific OBE for a particular US-APWR project site is set higher than 1/3 of the site-specific SSE, then explicit analysis of OBE is required for purposes of equipment qualification. As discussed in Section 7.6 of IEEE Std 344-2004, it may be acceptable to consider fewer than five site-specific OBE events, provided that technical justification is provided.

As per the US-APWR equipment seismic qualification EQ programmatic requirements, it is the responsibility of MHI/MNES to ensure that appropriate OBE ISRS are identified and transmitted to equipment vendors/suppliers for their use in seismic qualification of their equipment.

B.8 Application of RRS and Equipment Qualification Criteria

The RRS, in conjunction with the following equipment qualification criteria established for the equipment seismic qualification program, serve as the basic inputs and technical requirements for analysis and testing of mechanical, electrical, and instrumentation equipment requiring seismic and dynamic qualification as per the MHI US-APWR EQ Program:

- Seismic Qualification of Category I Mechanical Equipment and Inline Fluid System Components (including inline mounted equipment and supports)
- Seismic Qualification of Category I (Class 1E), Electrical and Instrumentation Equipment (including supports)
- Seismic Qualification of Category I Reservoirs and Tanks (including supports)
- Seismic Qualification of Category II Electrical and Mechanical Equipment (including supports)

The list of US-APWR Equipment Qualification Program Procedures provided in Attachment A and equipment criteria establish the requirements for seismic qualification of equipment and supports by analysis, testing, or a combination of analysis and testing, using the applicable ISRS. The requirements for qualification by analysis and qualification by testing are discussed in sections 4.10.10 and 4.10.11 respectively.

The US-APWR EQ Program procedures previously discussed provide specific programmatic direction on how to apply and implement the ISRS and the criteria discussed above during the design and procurement phases of a US-APWR project. The procedures direct that the criteria are to be referenced by or attached to procurement packages for mechanical, electrical, and instrumentation equipment that is required to be seismically qualified. As per the project procedures, vendors/suppliers of equipment for the US-APWR project are required to demonstrate conformance with the criteria. The qualification criteria may also be used by equipment manufacturers to establish and substantiate performance claims and verify equipment performance as part of their overall qualification effort. As per the US-APWR EQ Program directives and procedures, any conflicts between the criteria and the equipment purchase specification, equipment design specification, or the codes and standards must be brought to the attention of the PEQO.

The following US-APWR seismic design criteria for equipment-related commodities and distribution systems complement the equipment qualification criteria of the equipment seismic qualification program:

- Heating, Ventilation and Air Conditioning (HVAC) Duct and Duct Supports Design Criteria
- Seismic Qualification of Cable Trays and Supports Design Criteria
- Conduit and Conduit Support Design Criteria

B.9 Vendor Certification Using Previous Qualification Data or Comparison by Similarity

The US-APWR Equipment Qualification Program allows for equipment to be seismically qualified by the vendor using the results of previous analysis, generic (type) testing, or previous testing for another nuclear plant project-specific application. In these cases, US-

APWR Equipment Qualification Program procedures and criteria allow for seismic qualification by demonstrating that applicable US-APWR ISRS are enveloped by the RRS used for the previous qualification, and provided that all other US-APWR programmatic and technical requirements for qualification are met.

B.10 Qualification by Analysis

Analysis under the scope of the US-APWR program for seismic qualification of equipment conforms to the requirements for seismic analysis established in Tier 2 Chapter 3 Section 3.7 of the US-APWR DCD. The requirements for analytical modeling and the methods of seismic analysis defined in Tier 2 Chapter 3, Sections 3.7, 3.9, and 3.12 of the US-APWR DCD are adopted for the US-APWR program for seismic qualification. Requirements for qualification by analysis are specified in the equipment qualification criteria previously listed.

The analysis method is not recommended for complex equipment that cannot be modeled to correctly predict its response. Qualification performed by the analysis method is expected to be extensive enough to consider all critical details of a component or assembly. Analysis without testing is acceptable only if structural integrity alone can assure the design-intended function.

In cases of complex equipment, it may be acceptable to perform qualification by analysis on a portion or portions of a component or equipment assembly that can be accurately and reliably modeled, and to perform qualification by testing on the remaining portion(s). In these cases, the limits and interfaces of each method must be clearly explained and in sufficient detail to provide adequate justification for the approach used. The choice of applied seismic analysis method depends on the desired level of precision and the level of complexity of the particular equipment or component being qualified. Detailed descriptions of seismic analysis methods are contained in the DCD and are therefore are discussed only briefly here. The methods of analysis include:

- Modal response spectra analysis, which uses the broadened RRS as direct input for seismic qualification. For this method of analysis, the combination of multi-modal and multi-directional responses is in accordance with RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.”
- Time history analysis, which uses for seismic qualification input time histories generated from the broadened RRS. The generation of input time histories and the methodology of analysis should conform to the general requirements of NUREG 0800 SRP 3.7.1, unless justification is otherwise provided. The time history for each direction of the earthquake motion must be statistically independent of the others. The equipment / component responses for each orthogonal direction are to be combined using either Square Root Sum of the square (SRSS) or the Newmark 100%-40%-40% method in accordance with RG 1.92.
- Equivalent static load analysis. The equivalent static load method of analysis is generally recommended for seismic qualification of rigid equipment or equipment support structures whose dynamic response can be represented by models with few degrees of freedom. The equivalent static load method is relatively simple and more conservative than the other more detailed methods. The US-APWR seismic qualification program adopts the same ISRS peak acceleration factors as documented in Subsection 3.7.3 of the US-APWR DCD. Maximum equipment/component responses for each orthogonal direction are to be combined using either SRSS or the Newmark 100%-40%-40% method in accordance with RG 1.92.

For the purpose of seismic and dynamic qualification of equipment by analysis, the rigid response range is defined as that having a natural frequency greater than 50 Hz. This is consistent with the CSDRS defined in Tier 2 Chapter 3 Subsection 3.7.1 of the US-APWR DCD. However, as clarification, for the purpose of testing equipment that is not sensitive to response levels caused by high frequency ground motions, rigid is defined as equipment with a natural frequency greater than 33 Hz. If the equipment to be tested is sensitive to response caused by high frequency ground motions, then rigid is defined as equipment having a natural frequency greater than 50 Hz. See the discussion of high-frequency exceedances of earthquake ground motion for further discussion.

Load combinations and load factors for analysis are specified in the equipment seismic qualification program criteria documents previously listed, and follow the load definitions and load combinations given in Chapter 3 Sections 3.8 and 3.9 of the US-APWR DCD.

Damping values used in qualification by analysis are specified in the qualification criteria documents and are consistent with those given in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and Table 3.7.3-1 of the US-APWR DCD, unless otherwise justified in the form of documented test data. MHI Technical Reports "Dynamic Analysis of the Coupled RCL-RB-PCCV-CIS Lumped Mass Stick Model, MUAP-08005, Mitsubishi Heavy Industries, Ltd., April 2008" and "Seismic Design and Analysis of Power Source Buildings, MUAP-08002, Mitsubishi Heavy Industries, Ltd., February 2008" provide ISRS for various damping values. If the ISRS contained in these documents contain spectra at damping values that do not match the damping values of the equipment being analyzed, the analysis may be performed using a conservative value of damping which is lower than the value anticipated for the equipment and matches one of the damping values that is available. Alternatively, the US-APWR equipment seismic qualification criteria provide guidance for calculating ISRS at intermediate damping values. Otherwise, US-APWR EQ program procedures require that MHI/MNES be contacted for resolution.

B.11 Qualification by Testing

The US-APWR program for seismic and dynamic qualification by testing conforms to the technical requirements of IEEE Std 344, incorporates pertinent seismic and dynamic requirements from IEEE Std 323 and ASME QME-1, and adopts additional limitations and restrictions imposed by RG 1.100, USNRC DG-1175, and RG 1.148. Requirements for seismic and dynamic qualification by testing are specified in the equipment qualification criteria previously listed and include direction for appropriate equipment mounting and input waveform (frequency content, amplitude, and duration to generate a response at any point in the equipment sufficient to adequately replicate the anticipated design motion and fatigue effects).

Acceptable testing methods include proof, generic (type), and fragility testing. In accordance with IEEE Std 323, the following basic requirements are applied to all seismic qualification by testing for the US-APWR EQ program:

- Testing must exhibit a 10% margin above those acceleration requirements established at the mounting point of equipment unless otherwise justified in the seismic qualification report (Reference Section 6.3.1.6 of IEEE Std 323).
- Prior to seismic qualification testing, equipment or devices must be aged to the end of their service life, including applicable effects of all other relevant environmental aging

mechanisms such as mechanical cycling (fatigue), radiation, temperature, pressure, humidity, chemical degradation, and synergistic effects.

As per seismic qualification program criteria, when qualification by testing is used, the Test Response Spectra (TRS) must envelope the RRS derived from the ISRS over the entire frequency range of interest, except where high-frequency exceedances exist (greater than 20 Hz) and the equipment has been demonstrated to be insensitive to high-frequency disturbances. It is also preferred that the damping value of the RRS be the same as that of the TRS. In cases where the equipment damping is not established, it is recommended that the TRS be performed at 5% damping. When the damping for the TRS is greater than that for the RRS and the test method criteria are satisfied, then the damping is considered acceptable since it will produce conservative results. As per IEEE Std 344, when the damping in the TRS is less than that in the RRS (for the frequency range of interest), a conclusive statement is not possible without further evaluation, including revised damping values for the TRS.

B.12 Qualification by Combined Testing and Analysis

The US-APWR program for seismic qualification permits individual equipment to be qualified using a combination of testing and analysis where it is not practical to perform qualification by testing or analysis alone. This is anticipated for large equipment such as motors, pumps, or multi-bay equipment racks and consoles that cannot fit on a shake table or has too large of a mass to be handled by a shake table. In these cases, modal testing may be employed to identify resonant frequencies and mode shapes for correlation with an analysis.

B.13 Qualification Using Experience Data

The US-APWR DCD Tier 2 Chapter 3 Subsection 3.10.1.1 "Qualification Standards" and Subsection 3.10.4.2 "Experience Based Qualification" state that "Experience-based qualification is not used for any equipment." Therefore, the US-APWR EQ Program does not permit use of an experience-based approach for equipment qualification.

Qualification of equipment using an experience-based approach involves qualification by comparison that justifies similarity with previously qualified equipment that has been exposed to more severe in-plant vibration or natural seismic disturbances. The experience-based approach is also commonly referred to as the seismic qualification utility group (SQUG) approach which was used for qualification of existing equipment in older nuclear power plants as part of the resolution for USNRC USI-46. The experience-based approach typically relies on a previously established database of either earthquake experience data or test experience data and is greatly dependent on the technical basis provided for justification of similarities.

Qualification of equipment using an experience-based approach is permitted by IEEE Std 344 subject to the limitations and restrictions imposed by USNRC DG-1175 and a case-by-case review of the USNRC and with respect to:

1. The credibility and completeness of the compilation of the experience database
2. The rules for inclusion/exclusion of equipment in the experience database
3. The justification used to demonstrate similarity among the member items in a reference equipment class and the similarity between equipment in the experience database and those in the US-APWR project to be qualified
4. The justification used to demonstrate the reference equipment class functionality

5. The credibility of similarity among member items of a reference equipment class if a generic reference equipment class is proposed.

Further, in accordance with USNRC DG-1175, the experience-based approach (earthquake or test experience data) shall not be used for seismic qualification of electrical and active mechanical equipment that is exposed to harsh environments, aging, and earthquakes. Test experience data shall not be used for seismic qualification of high-frequency-sensitive equipment unless the tests were performed using IEEE Std 344-type tests with intentional high-frequency contents. Use of experience data (earthquake or test experience data) shall be avoided for seismic qualification of equipment identified in USNRC DG-1175 as:

1. Certain active electric components that may inadvertently change state during an earthquake such that they do not consistently perform their intended safety functions during and/or after an earthquake, such as certain types of relays, contactors, circuit breakers, switches, sensors, and potentiometers
2. Fragile electronic components, such as solid-state relays and microprocessors-based components
3. Electric equipment, such as battery chargers, inverters, relay and control panels, switchgear, and motor control centers (since the performance of this equipment is sensitive to its locations, orientations, and type of mounting within the plant).

In conclusion, an experience-based approach for equipment qualification shall not be implemented on any specific US-APWR project without an approved revision to or an approved departure from the DCD, and is subject to a case-by-case review by and the limitations imposed by the USNRC.

B.14 High-Frequency Exceedances of Earthquake Ground Motion

Historically, there have been occurrences of ground motions which have caused an exceedance of a plant's design spectra in the high frequency range. Based on nuclear plant operating experience, the high frequency response motion exceedances were found to be non-damaging to passive structural components, which are typically qualified by analysis. However, nuclear industry experience has found that certain SSCs, in particular fragile components such as relays, contactors, circuit breakers, switches, sensors, potentiometers, microprocessors-based components and other electrical and instrumentation and control devices whose output signals could be affected by high frequency excitation, are potentially sensitive and can be damaged by high frequency exceedances of the design spectra.

The US-APWR seismic qualification program adopts the guidance of IEEE Std 344, and USNRC DG-1175, USNRC "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion In Design Certification and Combined License Applications." to establish a process to identify, evaluate, and qualify or eliminate such SSCs that are potentially sensitive to high frequency exceedances.

Additional equipment evaluation by screening and subsequent qualification testing, depending on screening results, is required when ISRS used for equipment qualification exhibit high-frequency exceedances due to site-specific exceedances of the ground motion response spectra. As per the guidance of Section B.1 of USNRC DG-1175 and USNRC interim staff guidance, such evaluations must be performed when exceedances occur in the 20 – 50 Hz range, and must demonstrate both structural integrity and functionality for seismic category I equipment.

The components identified for high-frequency exceedance evaluation are consistent with those identified in USNRC DG-1175 and those historically identified in the U.S. nuclear industry. Those components are also consistent with EPRI white paper, "Seismic Screening of Components Sensitive to High Frequency Vibratory Motions", Palo Alto, California, June 2007. The detailed requirements for evaluation and qualification or elimination of SSCs that are potentially sensitive to high frequency exceedances are addressed in the equipment seismic qualification criteria previously listed.

B .15 Interfaces

US-APWR EQ Program procedures and directives ensure proper interfaces for implementation of seismic qualification criteria and requirements. For example, US-APWR EQ Program procedures ensure that:

- Equipment vendors/suppliers are suitably qualified to comply with US-APWR EQ Program requirements
- The appropriate seismic RRS (ISRS, GMRS, and FIRS, as applicable) are properly defined and transmitted to equipment vendors and suppliers for their use in equipment qualification
- In cases where it becomes apparent that equipment will be mounted at a location or elevation for which there are no corresponding ISRS, that the PEQO is contacted for resolution
- Equipment vendors/suppliers submit test procedures and methods, including proposed test table input motion, to the PEQO for review and approval prior to implementation
- Equipment vendors/suppliers submit all requisite documentation and records, including but not limited to properly formatted seismic qualification reports, to provide a complete demonstration of qualification for each piece of equipment, assembly, or component/device, in accordance with the US-APWR qualification criteria and programmatic requirements
- Overall equipment support reactions are included in the vendor/supplier seismic qualification report that is submitted to the PEQO in order that the equipment supporting structure (floor slab or beam, wall, etc.) and the equipment anchorage can be evaluated and/or designed
- Equipment assemblies and devices/components shall be mounted in the same configuration and orientation for which they were qualified, unless specific technical justification is otherwise included in the equipment qualification submittal
- Any conflicts between the qualification criteria and the equipment purchase specification, equipment design specification, or the codes and standards are brought to the attention of MHI/MNES for resolution
- Appropriate PEQO design and qualification reviews and approvals of vendor supplied equipment are performed