3.11 Environmental Qualification of Mechanical and Electrical Equipment

This section provides the U.S. EPR approach to the Environmental Qualification (EQ) of equipment and identifies the equipment that is within the scope of 10 CFR 50.49 including instrumentation and control (I&C) and certain accident monitoring equipment specified in RG 1.97. This section also addresses equipment that is capable of performing design safety functions under normal environmental conditions, containment test conditions, anticipated operational occurrences, accident, and post-accident environmental conditions.

The approach described in this section complies with the requirements of GDC 1, 2, 4, and 23; 10 CFR Part 50, Appendix B, Quality Assurance Criteria III, XI, and XVII; and 10 CFR 50.49.

Further discussion on compliance with the above GDC is provided below:

- GDC 1 requires that components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Components in the scope of this section that are subject to environmental design and qualification have auditable records to document that environmental design and qualification requirements have been met.

- GDC 2 requires that components important to safety are designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Components in the scope of this section that are subject to environmental design and qualification are designed with consideration of the environmental conditions or stressors resulting from natural phenomena as part of the environmental conditions evaluated. Additional information is provided in Section 3.2.

- GDC 4 requires that components important to safety are designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents (LOCA). Components in the scope of this section are protected against dynamic effects, including those of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Components in the scope of this section are also designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

- GDC 23 requires that protection systems are designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, pressure, steam, water, or radiation) are experienced. Components in the scope of this section that are subject to environmental design and qualification requirements are designed with consideration of the failure mode of the equipment.

The seismic qualification of mechanical and electrical equipment is presented in Section 3.10. The portions of post-accident monitoring equipment required to be environmentally qualified are discussed in Section 3.11.2.1.

A COL applicant that references the U.S. EPR design certification will maintain the equipment qualification test results and qualification status file during the equipment selection, procurement phase and throughout the installed life in the plant.

3.11.1 Equipment Identification and Environmental Conditions

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, core cooling, and containment and reactor heat removal, or are otherwise essential to preventing significant release of radioactive material to the environment.

Included in this equipment scope is:

- Equipment that performs these functions automatically.
- Equipment that is used by the operators to perform these functions manually.
- Equipment whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions.
- Safety-related and important to safety electrical equipment (including I&C) as described in 10 CFR 50.49 (b)(1) and (b)(2).
- Certain post-accident monitoring (PAM) equipment as described in 10 CFR 50.49(b)(3).

3.11.1.1 Equipment Identification

The list of components to be screened for qualification has been developed with consideration of systems, structures and components (SSC) located in three plant areas: the Nuclear Island (NI), Turbine Island (TI), and the balance of plant (BOP).

3.11.1.1.1 Nuclear Island

The NI consists of the following structures:

- Reactor Building (RB).
- Safeguards Buildings (SB).

- Fuel Building.
- Nuclear Auxiliary Building.
- Emergency Power Generating Building.
- Radioactive Waste Processing Building.
- Essential service water pump structure.
- Essential service water cooling tower structure.
- Vent stack.

The U.S. EPR is designed with a defense-in-depth concept through improvements of safety systems that include redundancy, separation features, and diversification of safety functions. For example, there are four SBs within the NI. Safety systems within the SBs are designed for redundancy with four trains and are located in physically separated divisions. SBs 1 and 4 are spatially separated on opposite sides of the RB; SBs 2 and 3 are housed together in a hardened enclosure.

Each of the four SB divisions is separated into two functional areas: mechanical and electrical. The electrical area also includes I&C and heating, ventilation, and air conditioning (HVAC). Specifically, SB Divisions 1 and 4 contain the safety injection system/residual heat removal system (SIS/RHRS) component cooling water system (CCWS), emergency feedwater system (EFWS), and severe accident heat removal system (SAHRS). SB Divisions 2 and 3 contain the main control room and technical support center, equipment for I&C and electrical systems for the NI, SB ventilation, and safety chilled water systems.

The SIS/RHRS design also includes defense-in-depth features. The SIS/RHRS is a four train system, and the individual trains are radially assigned to the reactor coolant system (RCS) loops to minimize the connection lengths to the RCS. The CCWS supplies SIS/RHRS heat exchangers with cooling water. The CCWS is installed near the connecting SIS/RHRS, but in a different radiation zone because the activity level of both systems is different. The CCWS is located in a second outer row around the RB in the radiological non-controlled area of the SB. The EFWS is also located in the mechanical, radiological non-controlled area of the SB.

The redundancy, separation, and diversification within the NI also provide a margin of safety for postulated failures because of environmental stressors. These features minimize the amount of equipment within the EQ program and are described below.

Safety Building Redundancy

The NI design includes redundancy and separation in the four SBs. If a break is postulated in one SB, that building function is considered lost and the building is isolated from the remaining buildings. The loss of function of this SB and the results of the break have no impact on the other three SBs that remain in a mild environment with respect to elevated temperature and pressure.

Fluid System Separation

High energy lines that connect the containment and the SBs carry post-accident radioactive fluid to the SBs. This arrangement can result in a radiation harsh zone in the SBs if a pipe ruptures in the containment. However, the plant layout vertically separates fluid systems from electrical and I&C equipment within each SB. Table 3.11-3—Equipment Distribution in SBs shows floor elevations and equipment spaces in the SBs where equipment is located and Figure 3.11-1—Harsh and Mild Zones in SBs identifies the areas that are radiation harsh in the SBs based on this equipment location. In these areas, concrete floors physically separate equipment providing protection for the electrical and I&C equipment. In SB Divisions 1 and 4, which contain steam and main feedwater isolation and main steam relief system equipment, the rooms are designed to isolate these high energy sources from the other SBs.

Equipment Separation

The equipment in the SBs is in a mild temperature and pressure environment based on the postulated break, and is consequently outside the scope of 10 CFR 50.49(c)(3), which states ". . . environmental qualification of electrical equipment important to safety located in a mild environment are not included within the scope of this section." Components located in the radiation harsh areas of the SBs, as shown in Figure 3.11-1, are qualified to radiation-only conditions. Section 3.11.1.2 describes the radiation thresholds for harsh and mild environments. Components located in the remaining areas of the SB are in environmental or radiation mild conditions and are not included in the 10 CFR 50.49 equipment list in Table 3.11-1—List of Environmentally Qualified Electrical/I&C Equipment. However, components within the SBs may require EQ due to electromagnetic compatibility (EMC) per RG 1.180, although they are in a mild environment. EMC is also addressed in detail in EPRI TR-102323, Revision 3 (Reference 1). Components from the SBs in this EQ category are listed in Table 3.11-1. Table 3.11-2—List of U.S. EPR Important to Safety Systems Screened for the EQ Program provides a listing of systems that are screened for inclusion in the EQ Program.

The RB is considered an environmental and radiation harsh area. Equipment in the RB, which is within the scope of 10 CFR 50.49, requires consideration for the principal types of environmental stressors (e.g., temperature, radiation, pressure, humidity, moisture, steam, water immersion, chemicals). The RB design also limits EQ through separation. The RB is an integrated structure consisting of an inner Containment Building, an outer Shield Building, and an annular space between the two buildings that separates them. The Containment Building is a post-tensioned concrete cylinder lined with steel. The Shield Building is a cylindrical reinforced concrete structure. The nuclear steam supply system (NSSS) is located in the Containment Building and high energy piping that traverses the annulus is routed in guard pipes. Components in the RB requiring EQ are listed in Table 3.11-1, under the Reactor Coolant System subheading.

3.11.1.1.2 Balance of Plant (BOP) and Turbine Island (TI)

The BOP consists of the following structures:

- Switchyard area.
- Offsite system transformer area.
- Auxiliary power transformer area.
- Generator transformer area.
- Demineralized water storage area.
- Structure for effluent disposal.
- Access building.
- Water treatment building.
- Cooling tower structure.
- Workshop/warehouse.
- Central gas supply building.
- Auxiliary boiler building.
- Office and staff amenities building.
- Security access facility.

The TI consists of the following structures:

- Switchgear building.
- Turbine building.

The U.S. EPR defense-in-depth concept, discussed above for the NI, also applies to the BOP and TI structures through the separation features designed into the plant layout. The BOP and TI equipment is protected and isolated from NSSS postulated large high energy line breaks (HELB) and radiation harsh environmental zones because the buildings housing these components are physically separate and are spatially separated from the RB by the SBs 2 and 3, which are hardened enclosures.

The equipment in the TI and BOP Buildings are considered to be in a mild temperature and pressure environment as well as a mild radiation environmental zone. Therefore, consistent with 10 CFR 50.49, only select TI and BOP electrical components (e.g., switchgear, motor control centers (MCC), transformers) that might be susceptible to electromagnetic compatibility (EMC) per RG 1.180 or non-safety-related equipment whose failure could prevent a safety function per 10 CFR 50.49(b)(2), are screened for inclusion in the EQ program.

3.11.1.1.3 Equipment Review and Screening

Equipment in the scope of the EQ program is identified within the master equipment list (MEL) and is qualified based on the guidelines provided in IEEE Std 323. The EQ Program equipment review for qualification uses a multi-step process. The first step in this process screens the SSCs to determine the equipment safety function, based on the plant safety analysis and the regulatory definition of safety-functions to identify the safety-related, Class 1E items. The next step determines and screens equipment that is not safety related, but whose failure could prevent the performance of a safety function. This equipment is designated as "NS-AQ" (i.e., non-safety, but having augmented quality). The third step screens and determines PAM equipment that is required in accordance with RG 1.97. The final step develops the EQ List based on the screening process and criteria of 10 CFR 50.49. As part of the overall determination, the MEL is subjected to the EQ Program screening process, so that qualification attributes are highlighted and identified, which results in the equipment being "screened in" and placed on the EQ List or being "screened out" and eliminated from the EQ List.

Table 3.11-1 includes a detailed listing by equipment tag number of electrical and I&C equipment located in an environmental harsh or radiation harsh environment that require qualification. Table 3.11-2—List of U.S. EPR Important to Safety Systems Screened for the EQ Program provides a listing of systems that are screened for inclusion in the EQ Program. See Section 3.11.2.2 for a description of Environmental Qualification of mechanical equipment.

A COL applicant that references the U.S. EPR design certification will identify additional site-specific components that need to be added to the environmental qualification list in Table 3.11-1.

3.11.1.2 Definition of Environmental Conditions

The environmental conditions include anticipated operational occurrences and normal, accident and post-accident environments due to design basis events (DBE). The environmental parameters (e.g., radiation, temperature, chemical spray, humidity [steam], pressure, flooding) applicable to the various environmental conditions in specific plant building and room locations are specified in Section 3D.5 of Appendix 3D and in the tables and figures provided in Appendix 3D.

Service conditions are the actual environmental, physical, mechanical, electrical, and process conditions experienced by equipment during service. Plant operation includes both normal and abnormal operations. Abnormal operation occurs during plant transients, system transients, or in conjunction with certain equipment or system failures.

Service condition environments fall into two general categories of harsh and mild environments.

- Harsh environments (H) are plant areas where the environmental conditions significantly exceed the normal design conditions as a direct result of a DBE. This excludes the seismic-related DBEs that are discussed in Section 3.10. Harsh environments include environmentally harsh environments and radiation harsh environments as discussed below:
  - An environmentally harsh environment is a location that is subject to a break in the RCS, steam, or other HELB piping that significantly alters the environmental parameters of temperature, pressure, humidity, and/or flooding. This includes a LOCA, main steam line break (MSLB) within the containment building (see Figures 3D-1 and 3D-2) and outside the containment in the feedwater valve compartment (see Figures 3D-3 and 3D-4) and main steam valve compartment (see Figures 3D-5 and 3D-6). Other HELB breaks outside of containment in areas such as the SBs and fuel building, while creating adverse environments (e.g., temperature, pressure, humidity, and/or flooding) does not require consideration as a harsh environment because of the independent and redundant safety trains.
  - A radiation harsh environment is a location inside or outside containment where the radiation levels exceed the following thresholds:
    - ≥1.0E04 Rads gamma for mechanical equipment including non-metallics or consumables (e.g., O-rings, seals, packing, gaskets, lube oil, diaphragms).
    - ≥1.0E03 Rads gamma for electrical or digital equipment.
- Mild environments (M) are plant areas where the environment at no time would be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. A mild location is essentially an area not subject to DBEs (excluding seismic events) and

whose radiation levels are less than or equal to the thresholds discussed above for mechanical and electrical equipment.

A new service condition for plant equipment, especially for digital I&C systems, is electromagnetic compatibility (EMC). Addressing EMC involves testing to show that critical equipment will not be adversely affected by electromagnetic interference (EMI) or Radio Frequency Interference (RFI) in the plant environment. EMC is addressed in the EQ program as a service condition that must be considered to address proper operation under adverse conditions for digital I&C equipment and is one of the screening criteria for the EQ list in Table 3.11-1.

3.11.1.3 Equipment Operability Times

Equipment required to be environmentally qualified has one or more of the following safety functions: reactor trip, engineered safeguards actuation, post-accident monitoring, or containment isolation. These safety functions are identified for applicable equipment. For each safety function identified a period of operability has been assigned as follows: immediate operability (2 hours), short term (24 hours), medium term (4 months), or long term (1 year).

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Qualification of Electrical Equipment

Electrical equipment, which includes I&C, contains components associated with systems that are essential to emergency reactor shutdown, containment isolation, core cooling, containment and reactor heat removal, or are otherwise essential to preventing significant release of radioactive material to the environment.

Included in this equipment scope is:

- Equipment that performs one or more of these functions automatically.
- Equipment that is used by operators to perform these functions manually.
- Equipment whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions.
- Other electrical equipment important to safety as described in 10 CFR 50.49(b)(1) and (2).
- Certain PAM equipment as described in 10 CFR 50.49(b)(3) and noted below.

The I&C design consists of four-fold redundancy from sensor to actuator:

- Physical separation in four divisions (SBs 1, 2, 3, and 4).
- Two functionally diverse subsystems.

- Two processing levels.
- Redundant power supplies for each cabinet.

Electrical equipment identified to be in a harsh location, as described in Section 3.11.1.1, will be environmentally qualified by type testing or type testing and analysis using the guidance of IEEE Std 323-2003[1] and related standards shown in Table 3.11-4—Summary Comparison of IEEE Endorsed Standards versus Latest IEEE Standards (References 4 through 14) and Table 3.11-5—Summary of IEEE Non-Endorsed Standards (References 2, 3, and 15 through 22). These related standards address other equipment specific IEEE qualification standards, such as IEEE Std 317 (electrical penetrations), IEEE Std 334 (motors), IEEE Std 344 (seismic), IEEE Std 382 (actuators), IEEE Std 383 (cables), IEEE Std 638 (transformers), IEEE Std 650 (chargers/inverters), and IEEE Std 1205 (aging).

The following RGs provide guidance for meeting the requirements of 10 CFR 50, Appendix A, General Design Criteria GDC 1, 2, 4 and 23; 10 CFR 50 Appendix B, Criterion III, XI, and XVII, to 10 CFR 50 and 10 CFR 50.49, and are used for qualification purposes for the EQ Program: RGs 1.9, 1.40, 1.63, 1.73, 1.89, 1.97, 1.100, 1.209, 1.131, 1.152, 1.156, 1.158, 1.180, and 1.209. A comparison of the related qualification standards and the associated RG that endorses them is provided in Table 3.11-4. Table 3.11-5 provides a summary of the related qualification standards that are not associated with a RG.

NUREG-0588, Revision 1 (Reference 23), also provides guidance for assessing the compliance of an environmental qualification program with 10 CFR 50.49. As noted in SRP 3.11, for future plants, RG 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 to enhance the guidance provided in RG 1.89.

PAM equipment is also environmentally qualified in accordance with Regulatory guide 1.97, Rev 4. The method used to identify and qualify this equipment is described in Section 7.5. The minimum list of PAM equipment, which is identified in Section 7.5 as potentially requiring operation in harsh environments, is also qualified according to the acceptance criteria of Section 3.11. PAM equipment is identified as Type A, B, C, D or E, according to RG 1.97, Rev 4 and Type A, B, C and D is environmentally qualified as required by 10 CFR 50.49 and the guidelines of Branch

1. Section 3.11.2.3 provides the justification for the use of the latest version of the IEEE standards referenced in this section that have not been endorsed by existing RGs. AREVA NP maintains the option to use current NRC-endorsed versions of the IEEE standards.

Technical Position (BTP) 7-10. Type E variables are not required to be environmentally qualified. BTP 7-10 states:

"For plants using Revision 4 of Regulatory Guide 1.97, accident monitoring equipment identified as Type A, B, or C in accordance with that guide should be environmentally qualified as required by 10 CFR 50.49. Type D variables should be environmentally qualified for the particular accident's postulated environment at the installed location in accordance with the plant's licensing basis. Licensees converting to Revision 4 or performing modifications based on Revision 4 may reference previously accepted alternatives as their basis for deviations from the environmental qualification criteria in Revision 4."

Table 3.11-1 and Table 3.10-1 provide a more extensive list of PAM equipment than the minimum PAM list provided in Section 7.5. This is as a result of the EQ and Seismic qualification systems screening process that identified additional components, as potentially supporting PAM instrumentation. These lists will be reconciled when the complete PAM list is developed, as explained in Section 7.5, and subsequently incorporated into the COL applicant's or holder's FSAR.

The acceptability of safety-related and important to safety electrical equipment located in a mild environment and not subject to 10 CFR 50.49 or EMC is demonstrated and maintained by use of the following types of programs:

- A periodic maintenance, inspection or replacement program based on sound engineering practice and recommendation of the equipment manufacturer, which is updated as required by the results of an equipment surveillance program.
- A periodic testing program used to verify operability of safety-related equipment within its performance specification requirements. System level testing of the type typically required by the plant technical specifications may be used.
- An equipment surveillance program that includes periodic inspections, analysis of equipment and component failures, and a review of the results of the preventive maintenance and periodic testing program.

3.11.2.2 Environmental Qualification of Mechanical Equipment

This section demonstrates that the U.S. EPR's approach to qualification of mechanical equipment is in accordance with the Standard Review Plan (SRP) 3.11 and GDC 4 and is based on methods developed and accepted by the NRC for the current fleet of operating reactors as described in South Texas Project, Units 1 and 2, Docket Nos. STN 50-498, STN 50-499, 10 CFR 50.59 Summary Report (Reference 24), Request for Additional Information on Elimination of EQ of Mechanical Components, South Texas Project, Units 1 and 2 (STP) (TAC Nos. M98912 and M98913) (Reference 25), Response to Request for Additional Information on Elimination of EQ of Mechanical Components (Reference 26), and Licensee's 10 CFR 50.59 Evaluation of Elimination of

EQ of Mechanical Components, South Texas Project, Units 1 and 2 (STP) (TAC Nos. M98912 and M98913) (Reference 27).

Mechanical equipment is inherently more rugged than electrical equipment and is less vulnerable to environmental conditions. Mechanical equipment is designed to operate under hostile process conditions. GDC 4 states, in part, that components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including loss-of-coolant accidents. Mechanical equipment is designed to comply with GDC 4 by verifying the ability of the components to perform their required safety functions when exposed to internal and external, normal and abnormal operating conditions, and when exposed to external postulated accident environments. The engineering design process and program evaluates both metallic and non-metallic components to meet environmental conditions (e.g., radiation, temperature, pressure) for safety-related and important to safety mechanical equipment. The combination of operating temperatures and pressures are compared to the design parameters of each component to confirm and demonstrate that design limits are not exceeded. The effects of radiation are considered in the evaluations. These evaluations constitute a normal and accident environmental analysis in accordance with GDC 4.

For mechanical equipment, the environmental design and qualification process focuses on the materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms). Equipment records are maintained, and these records include the results of tests and material analyses used as part of the environmental design and qualification process for each mechanical component. Engineering design specifications are generated and used in the procurement of equipment, components, and parts that are to be qualified. Under the procurement program, compliance with GDC 4 through the evaluation of non-metallic parts in mechanical components will be based on material evaluations and the form, fit, and function methodology used in an item equivalency evaluation. A listing of the mechanical non-metallic or consumable parts that require EQ are provided in Table 3.10-1.

The need to maintain a separate mechanical equipment qualification (MEQ) program for the U.S. EPR was determined to be redundant, considering current engineering design programs. This determination was based on the regulatory precedent from current operating reactors that originally maintained an MEQ program. The NRC has determined that engineering design, procurement, maintenance, and surveillance programs are acceptable to comply with the guidance in the SRP 3.11 (References 24, 25, 26, and 27) for MEQ. Accordingly, evaluations documented in a separate MEQ program are not necessary to justify compliance for safety-related mechanical components. Compliance with GDC 4 is initiated and maintained through the engineering design, procurement, maintenance, and surveillance programs. These plant programs include inspections, testing, analyses, repairs, and replacements.

The mechanical equipment qualification operational program identifies the plant programs that demonstrate mechanical equipment accommodates the effects of DBEs. The maintenance program includes maintenance, surveillance, and periodic testing of mechanical equipment. Under the maintenance program, routine monitoring of mechanical equipment is performed to identify and prevent age-related degradation of non-metallic parts. The program also verifies that the safety function of the mechanical equipment is maintained in normal, abnormal, and accident environments. Similarly, the procurement, maintenance, and surveillance programs maintain the equipment in sufficient operating condition and generate necessary corrective actions. This is based on documentation that includes vendor certification, design and purchase specifications for replacement parts, and material evaluations for replacement parts.

To verify the effectiveness of these programs to maintain compliance with GDC 4, the program data and records are required to be reviewed periodically in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI and other inspection, insitu test and monitoring programs. This process demonstrates that the equipment has not suffered any degradation, which may include the effects of thermal, radiation, and/or cyclic aging.

3.11.2.3 Justification for Using Latest IEEE Standards Not Endorsed by a RG

This section provides the description and justification for using the latest IEEE standards not endorsed by current RGs for the qualification of equipment. This justification does not preclude the use of versions of IEEE standards that are currently endorsed by RGs.

The IEEE has periodically updated the standards to incorporate evolutionary thinking and approaches of the nuclear industry with regard to equipment qualification. Table 3.11-4 provides a summary comparison of the current IEEE standards to be used for equipment qualification and the associated RGs and revision that endorse them. As shown in Table 3.11-4, a number of the later IEEE standards recommended for use on the U.S. EPR are not currently endorsed by the NRC. The following is a discussion of these non-endorsed IEEE standards and the justification for their use.

For an IEEE standard, the “R” just prior to the year means that the previously cited version of the standard was “Reaffirmed” in the later year shown. Reaffirmation is an approval process whereby the document is not changed, just agreed to be re-issued, as is. This reaffirmation is performed and noted because there is an IEEE requirement that standards be re-evaluated every 5 years to determine if a revision is deemed necessary. If a change is needed, then the document will be revised, and the year of revision is cited for the new document. If no changes are needed, then the document is cited with the date of latest publication followed by a notation that it was reaffirmed and the year of reaffirmation.

3.11.2.3.1 IEEE Std 317-1983/R2003, Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations

The first issuance of this document was cited as IEEE Std 317-1976. It was then revised and issued as IEEE Std 317-1983 (Revision of IEEE Std 317-1976), and subsequently noted as reaffirmed in 1988. The document was again reaffirmed in 2003 and cited as IEEE Std 317-1983 (R2003), noted as the revision to IEEE Std 317-1976.

The latest NRC endorsement was for the 1983 version of the standard, via RG 1.63, Revision 3. As shown above, the 1983 version has been reaffirmed in 2003, but not revised. It is reasonable to conclude that, pending a revision to the document, the NRC endorsement of the 1983 version would also apply to the 2003 version. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 317-2003 as the document to be used for qualification.

3.11.2.3.2 IEEE Std 323-2003, Standard for Qualifying Class 1E Equipment for Nuclear Power Generation Stations

IEEE Std 323-1983/2003 editions place more emphasis on the utility for periodic surveillance and maintenance than IEEE Std 323-1974, although the standard imposes no new requirements in this area. In addition, IEEE Std 323-1983/2003 editions clarify the utilization of margin during testing as applied to environmental transients by either adding the temperature and pressure margin to the postulated service condition profile or by applying the peak transient twice, but not applying both types of margin simultaneously.

IEEE Std 323-1983/2003 editions incorporate the knowledge and experience gained in the application of the 1974 edition and recognize elements of 10 CFR 50.49. For example, this edition contains a distinction consistent with 10 CFR 50.49 regarding qualification methods applicable to equipment located in mild and harsh environments. Equipment may be qualified to either the 1983/2003 or 1974 edition of IEEE Std 323 to meet the requirements of 10 CFR 50.49.

Because most existing test reports were based on IEEE Std 323-1974 requirements, AREVA NP will apply the following guidelines:

- Equipment certified to IEEE Std 323-1983/2003 requirements that is also certified to IEEE Std 323-1974 version of the test report is considered acceptable for use.
- Equipment certified to IEEE Std 323-1983/2003 that was subjected to a new type test would also have a revised test report to document the new testing. Certification for this material will reflect a later test report, and this report will require approval prior to use of the equipment.

Certification to IEEE standards alone is insufficient for 10 CFR 50.49 equipment. The vendor must also certify to the applicable test report.

The latest edition of the standard, IEEE Std 323-2003, is a clarification and more up to date qualification standard that incorporates the knowledge and experience gained in the application of earlier standards. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 323-2003 as the document to be used for qualification.

3.11.2.3.3 IEEE Std 334-2006, Standard for Qualifying Continuous-Duty Class 1E Motors for Nuclear Power Generating Stations

The original document was published as IEEE Std 334-1971. The 1971 version specified two accident transients with 15°F margin on peak temperature and 10 percent on pressure. The 1994 version allowed either two transients or application of margin. The 2006 version specifies one transient with 15°F margin. The 1971 and 1994 versions do not address condition monitoring, but the 2006 version does.

The 1971 and 1994 versions define formettes and motorettes, but give no explanation how to include them in qualification. The 2006 version addresses how to include these items into the qualification test program as test specimens. The 1971 and 1994 versions do not include loading versus thermal requirements during qualification test. The 2006 version requires evaluation of the worst-case loading in DBA (continuous run or start/stop).

The latest edition of the standard, IEEE Std 334-2006 is a clarification and more up-to-date qualification standard that incorporates the knowledge and experience gained in the application of earlier standards. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 334-2006 as the document to be used for qualification.

3.11.2.3.4 IEEE Std 344-2004, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

IEEE Std 344-2004 provides the recommended practices for seismic qualification of class 1E equipment. The following is a summary of a comparison of the various versions of this standard.

The IEEE Std 344-1971/1975 versions do not mention the Seismic Qualification Utility Group experience databases. The 1987 and 2004 versions discuss experience databases and how to apply operating experience to seismic qualification. Similarity for type testing is mentioned briefly in IEEE Std 1971/1975. Further discussion is given in IEEE Std 1987/2004. The IEEE Std 344-1971/1975 versions address uniaxial and biaxial excitation only. The 1987/2004 versions specify triaxial (preferred), then biaxial, then uniaxial and axial independence must be justified.

The IEEE Std 344-1971/1975 versions specify RMF or single frequency testing; 1987/2004 specifies RMF or RIM. Per application RMF can be supplemented with single frequency for peaks. The IEEE Std 1971/1975 versions specify static and dynamic analysis methods in general terms. The IEEE Std 344-1987/2004 versions specify numerous varieties of static and dynamic analyses with specific guidance. The IEEE Std 344-1971/1975 versions discuss only resonant search and modal testing. The IEEE Std 344-1987/2004 versions specify resonant search and modal testing and requirements to address resonances in testing to justify coupling. Transmissibility plots are required.

The IEEE Std 344-1971/1975 versions discuss the low impedance method and the exploratory tests used for qualification method selection. The IEEE Std 344-1987/2004 versions allow exploratory tests to be used as input for dynamic/static qualification analyses. The IEEE Std 344-1971/1975 versions defined “damping;” the 1987/2004 versions provide a method for calculating damping. The IEEE Std 344-1971/1975 versions define “seismic vibration.” The IEEE Std 1987/2004 versions define and differentiate between Seismic and Non-Seismic vibration. The IEEE Std 344-1971/1975 versions defined “ZPA;” the IEEE Std 1987/2004 versions provide a method for calculating ZPA.

The latest edition of the standard, IEEE Std 344-2004, is a clarification and more up-to-date qualification standard that incorporates the knowledge and experience gained in the application of earlier standards. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 344-2004 as the document to be used for qualification.

3.11.2.3.5 IEEE Std 382-2006, Standard for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Stations

The following discussion provides technical justification for the use of IEEE Std 382-2006 versus IEEE Std 382-1972, as endorsed by RG 1.73, Revision 0. A comparison of these documents is provided below:

- Documentation: The 2006 version requires additional configuration detail and specimen selection justification over the 1972 version and is considered to be more conservative.
- Type Testing: The 1972 version defines type testing and requires it, but provides no guidance or information on how to accomplish it. The 2006 version requires strict adherence to type test procedures and provides a definitive means to determine representative specimens to qualify a complete range of different equipment sizes. Therefore, the 2006 version is considered to be more conservative than the 1972 version.
- Test Sequence/Synergisms: Although synergisms were unknown in the 1972 version, test sequence was specified, in a manner similar to that provided by IEEE Std 323-1974, endorsed by RG 1.89 Revision 1. The 2006 version does account for synergisms, and requires the most severe test sequence to be followed, in accordance with IEEE Std 323-1974/2003. Therefore, there is no significant difference between the two versions.

- Margin: Margin was not addressed in the 1972 version. The 2006 version requires that margin be addressed, in accordance with IEEE Std 323-1974/2003. Therefore, the 2006 version is considered to be more conservative than the 1972 version.

- Functional Tests: These are stated only in very general terms in the 1972 version. The 2006 version contains specific requirements for the performance of functional testing during the qualification program, including the specific times during testing. Therefore, the 2006 version is considered to be more conservative than the 1972 version.

- Monitoring of Data: The 1972 version requires specific variable types to be monitored; the 2006 version requires additional data monitoring and provides examples for use. Therefore, both versions can be considered equally conservative.

- Aging: The 1972 version required thermal aging, radiation aging, vibration conditioning, and cycling for a specific number of times. The 2006 version also requires thermal aging, radiation aging, vibration conditioning, and cycling, but to more cycles than previously required. Therefore, both versions are similar, but the 2006 version is considered more conservative.

- DBA Transients: The 1972 version required two peak transients with an additional 15°F margin. The 2006 version corrected the peak transient requirement and the overall margin requirement. (Refer to discussion on “Margin,” above.)

- Seismic: The 1972 version required seismic testing to be in accordance with IEEE Std 344-1971, endorsed by RG 1.100, Revision 0, and specified random, multi-frequency (RMF) testing. The 2006 version also requires RMF testing, required input motion (RIM) testing, and increased documentation and compliance with IEEE Std 344-2004. Therefore, the 2006 version is considered to be more conservative than the 1972 version.

- Service Conditions: The 1972 version specified actual power, signal, and environmental conditions to be used for type testing; the 2006 version provided the methodology for the determination of service conditions, based on particular applications and classes of use. Because qualification could be performed on a generic basis, without regard for the end-use, either method is considered acceptable, and the 2006 version has been selected in order to be consistent with the overall qualification program.

As a result of the above discussions, AREVA NP believes that the 2006 version of IEEE Std 382 is more conservative than the 1972 version; therefore, AREVA NP believes it is acceptable to use the 2006 version of IEEE Std 382 as the document to be used for qualification.

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

The following is a summary of a comparison of the various versions of IEEE Std 383-2003:

- The IEEE Std 383-1974 version requires type test only, but 2003 version allows operation experience, ongoing qualification, and analysis. The 1974 version includes cable flame testing. In the IEEE Std 383-2003 version, flame testing has been moved to IEEE Std 1202-1991.
- The IEEE Std 383-1974 version includes cable connector qualification testing. In the IEEE Std 383-2003 version, this has been moved to IEEE Std 572-2006. The IEEE Std 383-1974 version presents a tabular test specimen selection guide that is replaced by a text version in the IEEE Std 383-2003 version.
- The latest edition of the standard, IEEE Std 383-2003, is a clarification and more up to date qualification standard that incorporates the knowledge and experience gained in the application of earlier standards. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 383-2003 as the document to be used for qualification.

3.11.2.3.7 IEEE Std 387-1995/R2007, Standard for Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating

For IEEE Std 387, the first issuance of this document was cited as IEEE Std 387-1984. It was then revised and issued as IEEE Std 387-1995 (Revision of IEEE Std 387-1984), and subsequently noted as reaffirmed in 2001. The document was again reaffirmed in 2007 and cited as IEEE Std 387-1995 (R2007), noted as the revision to IEEE Std 387-1984.

The latest NRC endorsement was for the 1995 version of the standard, via RG 1.9, Revision 4. As shown above, the 1995 version has been reaffirmed in 2007, but not revised. It is reasonable to conclude that, pending a revision to the document, the NRC endorsement of the 1995 version would also apply to the 2007 version. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 387-2007 as the document to be used for qualification.

3.11.2.3.8 IEEE Std 535-1986/R1994, Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating

For IEEE Std 535-1986 it was noted as reaffirmed in 1994 and is currently cited as IEEE Std 535-1986 (R1994).

The latest NRC endorsement is for the 1986 version of the standard, via RG 1.158, Revision 0. As shown above, the 1986 version has been reaffirmed in 1994, but not revised. It is reasonable to conclude that, pending a revision to the document, the NRC endorsement of the 1986 version would also apply to the 1994 version. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 535-1994 as the document to be used for qualification.

3.11.2.3.9 IEEE Std 572-2006, Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating

The following is a summary of a comparison of the various versions of IEEE Std 572-2006.

- The IEEE Std 572-1985 version addresses electrical power, control, and non-axial instrument connectors; the IEEE Std 572-2006 version addresses all electrical and instrumentation connectors. This is neither more nor less restrictive, but the applicability scope is slightly larger.
- The IEEE Std 572-1985 version requires the qualification test to address the mechanisms of aging and degradation specified by IEEE Std 323. The IEEE Std 572-2006 version requires that all degrading effects must be addressed including dielectric stress, connecting or disconnecting wear. Atypical connectors discussed.
- The latest edition of the standard, IEEE Std 572-2006, is a clarification and more up-to-date qualification standard that incorporates the knowledge and experience gained in the application of earlier standards. Therefore, AREVA NP believes that it is acceptable to use IEEE Std 572-2006 as the document to be used for qualification.

3.11.3 Qualification Test Results

The summaries and results of qualification tests for electrical equipment and components are documented in the Equipment Qualification Data Packages (EQDP). Appendix 3D, Attachment A provides an EQDP sample format. Qualification of electrical equipment and components in a mild location is based on certificates of conformance to the procurement specification and other specifics as noted in Section 3.11.2.1. The EQ program is fully described in Sections 3.11.1 and 3.11.2.

The summaries and results of seismic qualification tests for electrical and mechanical equipment and components in the harsh environment areas are documented in the Seismic Qualification Data Packages (SQDP). Appendix 3D, Attachment F provides a sample SQDP format. Mechanical equipment qualification is based on conformance with engineering and procurement documents and other specifics as noted in Section 3.11.2.

If the equipment qualification testing is incomplete at the time of the COL application, a COL applicant that references the U.S. EPR design certification will submit an implementation program, including milestones and completion dates, for NRC review and approval prior to installation of the applicable equipment.

3.11.4 Loss of Ventilation

The environments resulting from loss of HVAC equipment are typically slow temperature transients with resulting steady state conditions, which are not harsh by definition. Because the equipment operates well below the maximum stress level capability in its normal environment, it is unlikely that low level, short duration temperature excursions caused by loss of HVAC will result in the maximum stress level capability being exceeded. Temperature monitors through out the plant record any change in temperature over time. The corrective action program and the maintenance rule program provide adequate controls to initiate repairs when necessary. In addition, because of the slow nature of this temperature change, time is available for operator action to correct the environmental problem by re-establishing or improvising ventilation. Because several means are available to the operator to correct the problem, the duration of the transient condition is expected to be short.

The normal and abnormal environmental conditions shown on Tables 3D-4 and 3D-7 reflect anticipated normal and maximum conditions based on loss of normal ventilation systems. Additional information on the U.S. EPR ventilation systems is provided in Section 9.4.

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environments

Applicable chemical environments are defined in Tables 3D-1, 3D-4, and 3D-7 for normal and abnormal operating conditions. Chemical spray is not considered for the U.S. EPR because chemical spray is not used to mitigate a DBE. For beyond DBEs, a SAHRS system is used to prevent the pressure and temperature within the containment from exceeding design limits. However, given the deliberate steps required to initiate and actuate the SAHRS system, inadvertent actuation of this system is not considered a credible event (Refer to Section 19.2.3.3.3.2).

The use of chemicals for pH control and their effects have been considered. Containment sump pH is adjusted to the range of 7.0 and higher if the containment is flooded. Information on pH adjustment is provide in Section 6.3, Containment pH Control, and is based on maintaining the Cesium Iodide (CsI) in the IRWST solution as discussed in Section 15.0.3.

3.11.5.2 Radiation Environments

Radiation environments are defined in Tables 3D-1, 3D-3, and 3D-8 for normal and accident conditions.

Normal operation radiation doses are calculated for initial plant start-up conditions. Radiation doses are continuously monitored during plant operation. If the actual measured radiation doses are higher than the original calculated doses, the U.S. EPR database will be revised and qualified life adjustments identified through the EQ program. The assumptions associated with the normal operations dose rates are discussed in Section 12.3.

The normal operations dose rates for equipment qualification are derived from direct gamma emitted by components that contain radioactive fluids. Because the structural walls of these components shield beta particles, beta radiation is omitted. Bremstrulung radiation is also neglected because it is a small contributor compared to the normal operations source term gamma contribution. The LOCA accident dose rates include a submersion dose and a direct dose contribution. The submersion dose is primarily from ESF leakage; the dose contribution is derived from both the gamma and beta radiation. The beta radiation may be attenuated by low-density equipment enclosures. Alpha radiation is neglected from both the normal and accident equipment qualification dose rates because the alpha particle is easily attenuated by air, and it is primarily a personnel committed dose concern

When area doses exceed the qualified dose of an item of interest, a component specific dose calculation may be performed to determine doses at the specific equipment location. The accident dose rates were calculated based on the alternate source term methodology. The assumptions associated with the accident dose rates are discussed in Section 15.0.3. See also the discussion in Section 3D.5.51.

3.11.6 Qualification of Mechanical Equipment

See Section 3.11.2.2.

3.11.7 References

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

2. IEEE Std C37.82-1987/R2004, "IEEE Standard for the Qualification of Switchgear Assemblies for Class 1E Applications in Nuclear Power Generating Stations."

3. IEEE Std C37.105-1987/R1999, "IEEE Standard for Qualifying Class 1E Protective Relays and Auxiliaries for Nuclear Power Generating Stations."

4. IEEE Std 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations."

5. IEEE Std 317-1983/R2003, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations."

6. IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

7. IEEE Std 334-2006, "IEEE Standard for Qualifying Continuous-Duty Class E Motors for Nuclear Power Generating Stations."

8. IEEE Std 344-2004, "IEEE Recommended Practices for Seismic Qualification of Cass 1E Equipment for Nuclear Power Generating Stations."

9. IEEE Std 382-2006, "IEEE Standard for Qualification of Safety Related Actuators for Nuclear Power Generating Stations."

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

11. IEEE Std 387-1995/R2007, "IEEE Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations."

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

13. IEEE Std 535-1986/R1994, "IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations."

14. IEEE Std 572-2006, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations."

15. IEEE Std 627-1980/R1991, "IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Station."

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

17. IEEE Std 638-2006, "IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Station."

18. IEEE Std 649-2006, "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations."

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

20. IEEE Std 1202-1991/R1996, "IEEE Standard for Flame Propagation Testing of Wire and Cable."

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

22. IEEE Std 1290-1996/R2005, "IEEE Guide for Motor Operated Valve (MOV) Motor Application, Protection, Control, and Testing in Nuclear Power Generation Station."

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

24. 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."

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

26. 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 EQ of Mechanical Components, “ Docket Nos. STN 50-498, STN 50-499, Units 1 and 2 (STP).

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