KHNPDCDRAIsPEm Resource

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Sent: Monday, June 22, 2015 8:22 AM
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Cc: Zhang, Deanna; Jackson, Terry; Ward, William; Lee, Samuel
Subject: APR1400 Design Certification Application RAI 43-7887 (07.01 - Instrumentation and Controls - Introduction)
Attachments: APR1400 DC RAI 43 ICE 7887.pdf; image001.jpg

KHNP

The attachment contains the subject request for additional information (RAI). This RAI was sent to you in draft form. Your licensing review schedule assumes technically correct and complete responses within 30 days of receipt of RAIs. However, KHNP requests, and we grant, the following days to respond to the RAI's questions. We may adjust the schedule accordingly.

07.01-10: 45 days
07.01-11 : 45 days
07.01-12 : 60 days
07.01-13 : 60 days
07.01-14 : 60 days
07.01-15 : 90 days
07.01-16 : 45 days
07.01-17 : 60 days
07.01-18 : 60 days
07.01-19 : 60 days
07.01-20 : 90 days
07.01-21 : 90 days
07.01-22 : 30 days
07.01-23 : 90 days
07.01-24 : 90 days
07.01-25 : 30 days

Please submit your RAI response to the NRC Document Control Desk.

Thank you,

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

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REQUEST FOR ADDITIONAL INFORMATION 43-7887

Issue Date: 06/22/2015
Application Title: APR1400 Design Certification Review – 52-046
Operating Company: Korea Hydro & Nuclear Power Co. Ltd.
Docket No. 52-046
Review Section: 07.01 - Instrumentation and Controls - Introduction
Application Section: Section 7.1

QUESTIONS

07.01-10
Apply the correct reference to 10 CFR 50.54(jj) and 50.55(i).

10 CFR 50.54(jj) and 10 CFR 50.55(i) state that structures, systems, and components subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety function to be performed. This requirement was moved from 10 CFR 50.55a(a)(1) in November 2014 (79 FR 65776). APR1400 Final Safety Analysis Report (FSAR) Tier 2, Section 7.1.2.2, references 10 CFR 50.55a(a)(1) instead of 10 CFR 50.54(jj) and 10 CFR 50.55(i). Modify the APR1400 FSAR to reflect the change in regulations.

07.01-11
Clarify whether the applicable I&C systems in Table 7.1-1 meet the requirements of 10 CFR 50.54(jj) and 10 CFR 50.55(i). In addition, demonstrate how the requirements of 10 CFR 50.54(jj) and 10 CFR 50.55(i) are met.

10 CFR 50.54(jj) and 10 CFR 50.55(i) state that structures, systems, and components subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety function to be performed. This requirement was recently moved from 10 CFR 50.55a(a)(1). Tier 2, Section 7.1.2.2, of the APR1400 FSAR states that the "The I&C [instrumentation and controls] systems that are applicable to 10 CFR 50.55a(a)(1) (Reference 8), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.55a(a)(1) by complying with IEEE Std. 603 (Reference 9), Clause 5.3." This description does not clearly state that the I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.54(jj). Clarify whether the intent of this statement is "The applicable I&C systems listed in Table 7.1-1 are designed to meet the requirements of 10 CFR 50.54(jj) and 10 CFR 50.55 (i). These systems meet the requirements of 10 CFR 50.54(jj) and 10 CFR 50.55(i) by complying with the requirements of IEEE Std. 603 (Reference 9), Clause 5.3." Further, the applicant does not provide a reference on how the requirements of IEEE Std. 603-1991, Clause 5.3 are met. Provide a reference to where compliance to IEEE Std. 603-1991, Clause 5.3 is discussed in the application. Modify the FSAR to include this information.

REQUEST FOR ADDITIONAL INFORMATION 43-7887

07.01-12
Clarify whether the applicable I&C systems in Table 7.1-1 meet applicable NRC regulations in order to meet the requirements of 10 CFR 52.47a(2) and 10 CFR 52.47a(3)(i).

10 CFR 52.47a(2) requires applicants to provide a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. 10 CFR 52.47a(3)(i) requires applicants to provide information on "The principal design criteria for the facility. Appendix A to 10 CFR Part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units."

APR1400 FSAR, Tier 2, Sections 7.1.2.2 thru 7.1.2.35 identify regulations that the APR1400 I&C systems are designed in accordance to. This description does not clearly state that these I&C systems meet the requirements of these NRC regulations. For example, APR1400 FSAR, Tier 2, Section 7.1.2.5 states that "The I&C systems that are applicable to 10 CFR 50.34f(2)(v) (Reference 12), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(v)." The applicant should state that the applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(v). Modify APR1400 FSAR, Tier 2 Sections 7.1.2.2 thru 7.1.2.35 to clearly state whether the I&C systems meet the requirements of NRC regulations. Ensure that the references are properly stated [e.g. 50.34(f) vice 50.34f].

07.01-13
Provide reference to sections in the APR1400 FSAR that contain information on how NRC regulations are met in order to meet 10 CFR 52.47a(2) and 10 CFR 52.47a(3)(i).

10 CFR 52.47a(2) requires applicants to provide a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. 10 CFR 52.47a(3)(i) requires applicants to provide information on the "principal design criteria for the facility. Appendix A to 10 CFR Part 50, General Design Criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units."

APR1400 FSAR, Tier 2, Section 7.1.2, "Identification of Safety Criteria" identifies safety regulations that the APR1400 I&C systems are designed in accordance to. For several of the regulations (i.e. 10 CFR 50.34f(2)(xx), 10 CFR 50.55a(h)(3) which requires compliance to IEEE Std. 603-1991, GDC 10, GDC 13, GDC 15, GDC 16, GDC 24, GDC 25, GDC 28, GDC 29, GDC 33, GDC 34, GDC 35, GDC 38, GDC 41, and GDC 44), the applicant did not include references to applicable APR1400 FSAR sections and technical reports that contain information on how these regulations are met. Modify the applicable APR1400 FSAR, Tier 2, sections to include the appropriate references.

07.01-14
Describe how the requirements of 10 CFR Part 50, Appendix A, GDC 21 relate to IEEE Std. 603-1991 requirements and demonstrate how both of these requirements are met in the APR1400 design.

GDC 21 states “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.”

Tier 2, Section 7.1.2.23, of the APR1400 FSAR states “The I&C systems that are applicable to GDC 21, as shown in Table 7.1-1, are designed in accordance with GDC 21. The protection system is designed to comply with the requirements of IEEE Std. 603. No credible single failure would result in a loss of the protection function.” As is written, it appears that the applicant is trying to relate the requirements of GDC 21 to IEEE Std. 603. However, the applicant does not describe how the two requirements relate to each other (i.e. specify the specific clauses of IEEE Std. 603-1991 that map to the requirements of GDC 21) with respect to the APR1400 design. In addition, GDC 21 provides requirements that are not found in IEEE Std. 603 (e.g., capability to perform safety function with a single failure and with a component/channel out-of-service). Modify the FSAR to include this information.

07.01-15
Describe the design techniques that will be used to prevent loss of the protection function in order to meet the requirements of 10 CFR Part 50, Appendix A, GDC 22 are met.

GDC 22, "Protection system independence" states, “The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.” The APR1400 FSAR, Tier 2, Section 7.1.2.24, "Conformance with GDC 22" states, “The I&C systems that are applicable to GDC 22, as shown in Table 7.1-1, are designed in accordance with GDC 22. The protection systems comply with the independence requirements of IEEE Std. 603 except for the CEA [control element assembly] position inputs described in Subsection 7.1.2.3.” The applicant does not describe the design techniques (e.g. functional diversity and other design techniques) that will be used to prevent loss of the protection function. Modify the FSAR to include this information.

07.01-16
Clarify how the Plant Protection System (PPS) and Engineered Safety Features Actuation System (ESFAS) address the guidance of Staff Requirements Memorandum (SRM) to SECY-93-087, Item II.Q.

SRM-SECY-93-087, Item II.Q. requires the applicant to demonstrate that vulnerabilities to software common cause failures in the safety system are adequately addressed. Tier 2, Section 7.1.2.36, of the APR1400 FSAR states “Analyses and design features for diversity and defense-in-depth for the PPS and ESFAS are provided in accordance with SECY-93-087, Item II.Q (Reference 23), as referenced by NUREG-0800 (Reference 24).” This statement does not reference the SRM to this SECY which is the

Commission's position on diversity and defense-in-depth for computer-based systems. Clarify in the FSAR how the APR1400 I&C design conforms to the SRM to SECY-93-087, Item II.Q.

07.01-17

Clarify how the alarm systems conform to the guidance of the SRM to SECY-93-087 Item II.T.

SRM-SECY-93-07 Item II.T. states that the alarm system for advanced light water reactors should meet the applicable EPRI requirements for redundancy, independence, and separation. In addition, alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits. APR1400 FSAR, Tier 2, Section 7.1.2.37 states "The alarm systems are required to meet the redundancy, independence, and safety alarm system requirements in accordance with SECY-93-087, Item II.T (Reference 5)." This statement does not reference the SRM to this SECY which is the Commission's position on alarm systems. Clarify in the APR1400 FSAR how the alarm systems design conforms to the SRM to SECY-93-087, Item II.T.

07.01-18

Describe how the ex-core neutron flux monitoring system (ENFMS), auxiliary process cabinet-safety (APC-S), and the safety portion of radiation monitoring system (RMS) meet the requirements of IEEE Std. 603-1991, including Clauses 5.1, 5.3, 5.5, and 5.6. In addition, the applicant should clarify whether there are any other standalone safety-related I&C systems

10 CFR 50.55a(h)(3) states, in part, that application filed on or after May 13, 1999, for design certifications must meet the requirments for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. Clause 5.1, "Single-Failure Criterion," of IEEE Std. 603-1991 states, in part, "The safety systems shall perform all safety functions required for a design basis event (DBE) in the presence of: (1) any single detectable failure within the safety systems concurrent with all identifiable but non-detectable failures; (2) all failures caused by the single failure; and (3) all failures and spurious system actions that cause or are caused by the DBE requiring the safety functions. The single-failure criterion applies to the safety systems whether control is by automatic or manual means." Clause 5.3, "Quality," of IEEE Std. 603-1991 requires components and modules to be of a quality that is consistent with minimum maintenance requirements and low failure rates. Safety system equipment shall be designed, manufactured, inspected, installed, tested, operated, and maintained in accordance with a pre-scribed quality assurance program. Clause 5.5, "System Integrity," of IEEE Std. 603-1991 states that "The safety systems shall be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis." In addition, Clause 5.6, "Independence," of IEEE Std. 603-1991 requires independence between redundant portions of a safety system and between safety and non-safety systems.

APR1400 FSAR, Tier 2, Section 7.1 states "The following safety I&C systems are implemented on independent platforms that are diverse from the safety-qualified PLC platform: ENFMS (see Subsection 7.2.1.1.c), APC-S (see Subsection 7.2.1), safety portion of RMS (refer to Section 11.5 and Subsection 12.3.4)... " Design descriptions were not provided in the FSAR for the ENFMS, APC-S, and safety portion of the RMS to demonstrate that the requirements of IEEE Std. 603-1991, Clauses 5.1, 5.3, 5.5, and 5.6 are met. In addition, APR1400 FSAR, Tier 2, Table 7.1-1, "Regulatory Requirements Applicability Matrix," does not include the RMS. Furthermore, it is not clear whether there are any other standalone safety I&C systems besides the safety-related portion of the RMS. Identify all standalone, safety-related I&C

systems and demonstrate how these systems meet the applicable regulations (e.g. 10 CFR 50.54 (jj), 10 CFR 50.55(i), IEEE Std. 603-1991, etc.). Modify the FSAR to include this information.

07.01-19

Describe the turbine I&C system and how it interfaces with the safety-related I&C systems to meet the requirements of 10 CFR Part 50, Appendix A, GDC 1, GDC 24, and IEEE Std. 603-1991, Clause 5.6.3.

GDC 1, "Quality Standards and records" requires, in part, that “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.” GDC 24, "Separation of protection and control systems" states that, “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.” In addition, 10 CFR 50.55a(h)(3) states, in part, that application filed on or after May 13, 1999, for design certifications must meet the requirements for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. Clause 5.6.3 of IEEE Std. 603-1991 requires independence between safety and non-safety systems.

APR1400 FSAR, Tier 2, Section 7.1, states, “…independent systems such as the turbine/generator (T/G) control and protection system, the nuclear steam supply system (NSSS) monitoring system, and the balance of plant (BOP) monitoring system perform the required functions of a portion of the I&C systems.” The staff reviewed Chapter 10, "Steam and Power Conversion System," of the APR1400 FSAR, Tier 2, and could not find information on the design of the T/G control and protection system, including how this system interfaces with safety-related I&C systems to meet the independence requirements of GDC 24 and IEEE Std. 603-1991, Clause 5.6.3. Provide information on the design of the T/G I&C system and the interfaces of this system to safety-related I&C systems (e.g. plant protection system) in order to demonstrate compliance to GDC 1, 24, and IEEE Std. 603-1991, Clause 5.6.3. In addition, the applicant should clarify whether there are any other non-safety, standalone I&C systems that have interfaces to the safety-related I&C systems. If so, how do these standalone systems meet the requirements of GDC 24 and IEEE Std. 603-1991, Clause 5.6?

07.01-20

Verify that identification requirements for safety I&C systems conform to the guidance of RG 1.75 in order to demonstrate compliance to IEEE Std. 603-1991, Clause 5.11.

10 CFR 50.55a(h)(3) states, in part, that application filed on or after May 13, 1999, for design certifications must meet the requirements for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. IEEE Std. 603-1991, Clause 5.11, "Identification," requires that (1) safety system equipment be distinctly identified for each redundant portion of a safety system in accordance with the requirements of IEEE Std. 384-1981, (2) components or modules mounted in equipment or

assemblies that are clearly identified as being in a single redundant portion of a safety system do not themselves require identification, (3) identification of safety system equipment be distinguishable from other purposes, (4) identification of safety system equipment does not require frequent use of reference material, and (5) the associated documentation be distinctly identified in accordance with the requirements of IEEE Std. 494-1974. SRP Section 7.1, Appendix 7.1-C, provides staff review criteria on meeting the requirements of IEEE Std. 603-1991. Section 5.11 of this appendix states, “Guidance on identification is provided in Regulatory Guide (RG) 1.75, which endorses IEEE Std. 384-1992. The preferred identification method is color coding of components, cables, and cabinets.”

Section A.5.11, "Identification," of Technical Report APR1400 Z-J-NR-14001-P, Revision 0, "Safety I&C System," Appendix A, "Conformance to IEEE STD. 603-1991," states that all equipment, including panels, modules, and cables associated with the RPS [reactor protection system] and ESF [engineered safety features] systems, are marked in order to facilitate identification. The safety I&C system is configured in accordance with specific identification requirements which provide a standardized method for identifying equipment, diagrams and signals for the purpose of consistency during the installation process." The staff finds that the application did not state that these specific identification requirements of the safety I&C system will conform to the guidance of the guidance of RG 1.75. Verify whether these specific identification requirements conform to the guidance of RG 1.75 or if an alternate method is proposed. If an alternate method is used, provide justification as to why the method provides a comparable level of safety to the guidance in RG 1.75.

07.01-21

Identify and describe auxiliary features in the APR1400 design in order to demonstrate compliance to IEEE Std. 603-1991, Clause 5.12. Specifically, describe how auxiliary features within the protection system will not adversely impact safety functions. In addition, provide a comprehensive list of all safety-related auxiliary supporting features.

10 CFR 50.55a(h)(3) requires compliance to IEEE Std 603-1991. IEEE Std. 603-1991, Clause 5.12, "Auxiliary Features," states that (1) auxiliary supporting features shall meet all requirements of this standard, and (2) other auxiliary features that perform a function that is not required for the safety systems to accomplish their safety functions, and are part of the safety system by association, shall be designed to meet those criteria necessary to ensure that these components, equipment, and systems do not degrade the safety systems below an acceptable level. APR1400 FSAR Tier 2, Section 7.1.1.10, "Auxiliary Support Features" states that auxiliary supporting features and other auxiliary features are safety systems or components of systems that provide the services that are required for the safety systems to accomplish their safety functions. HVAC and electrical power systems are examples of auxiliary supporting features. The I&C aspects of auxiliary supporting features are described primarily in Chapters 8 and 9. Examples of other auxiliary features are built-in test equipment and isolation devices. Section A.5.12, "Auxiliary Features," of Technical Report APR1400-Z-J-NR-14001-P, Rev. 0, "Safety I&C System" states that any features (components, equipment and systems) of the safety I&C system that perform safety functions satisfy the Clause 5.12 requirements of IEEE Std. 603-1991. All of these features are designated as safety-related and are part of the safety I&C system. The communication architecture provides the ability to transmit information to non-safety related devices and is classified as safety-related until the non-safety boundary. Auxiliary features (bypass, control element assembly withdrawal prohibit (CWP) signal, test, and calibration functions) are designed not to affect the protection system from accomplishing their safety functions.

The staff finds that additional information is required regarding the auxiliary features that are designed to not affect the protection system from accomplishing their safety function. Specifically, the staff requests the applicant to identify and describe how the auxiliary features (bypass, CWP signal, test, and calibration functions) within the protection system do not affect the protection system from accomplishing their safety

functions. In addition, the application states that the HVAC and electrical power systems, built-in test equipment, and isolation devices are examples of safety-related auxiliary support systems. The staff requests the applicant to provide a comprehensive list of all safety-related auxiliary supporting features in the APR1400 design in order to meet the requirements of IEEE Std. 603-1991, Clause 5.12.

07.01-22

Remove from the APR1400 FSAR, Tier 2, Section 7.1.2.3, the statement “The I&C systems that are applicable to 10 CFR 50.55a(h)(2) (Reference 10), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.55a(h)(2) except that the CPCS [Core Protection Calculator System] has two channels of a reed switch position transmitter (RSPT) for each control element assembly."

10 CFR 50.55a(h)(2) states “For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603–1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.” 10 CFR 50.55a(h)(2) does not apply to current applications. Therefore, remove the reference to this requirement from the FSAR and modify the discussion on CPCS to the discussion on compliance to 10 CFR 50.55a(h)(3).

07.01-23

Clarify what standards are used for equipment qualification (EQ) of the safety-related I&C system equipment.

GDC 4, "Environmental and Dynamic Effects Design Bases" states, in part, that "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." RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference [EMI/RFI] in Safety-Related Instrumentation and Control Systems" states that it is intended that either set of test methods (MIL-Std-461E or IEC 61000-6-4) be applied in its entirety, without selective application of individual methods (i.e., no mixing and matching of test methods) for emissions testing. Technical Report APR1400-Z-J-NR-14001-P, Rev. 0, "Safety I&C System," Section 6.3, "EMI/RFI Testing," identifies both MIL-Std-461E and IEC-61000 Part 4 Series standards as used for EQ of safety-related I&C system equipment. Clarify which set of standards is used to demonstrate that the safety I&C system equipment will meet the EQ requirements for the electromagnetic compatibility or provide justification for why the sets of standards are mixed, which is contrary to the guidance in RG 1.180. If IEC-61000 Part 4 series standards are used, identify which version of those standards are used.

07.01-24

Clarify what qualification tests will be included in the EQ program of the safety-related I&C system equipment. Also clarify the inconsistency on which version for IEEE Std. 323 is used.

GDC 4 states, in part, that "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." RG 1.180, RG 1.100, RG 1.209 and their endorsed IEEE Std. 323-2003, EPRI TR-107330, and EPRI TR-102323, state, in part, that tests for electrical fast transient, electrostatic discharge, surge withstand capability, and Class 1E to Non-Class 1E isolation should be included as part of the EQ testing program for the safety I&C system equipment. The staff could not identify design information for these EQ tests in Section 6, “Equipment Qualification” of Technical Report APR1400-Z-J-NR-14001-P, Rev. 0, "Safety I&C System." Provide necessary design information to demonstrate that the EQ program of the safety I&C system equipment will include those EQ tests. Also clarify which associated ITAAC items will include those EQ tests. In addition, IEEE Std. 323-2003 is endorsed in RG 1.209. However, the 1983 and 2003 version of IEEE Std. 323 are used in Section 6 and Section A.5.4 of the above Safety I&C System TeR. Clarify the inconsistent use of version for IEEE Std. 323 and provide necessary justification for use of the older version of this IEEE standard.

07.01-25

Provide adequate design information and accompanying analysis to demonstrate predictable and repeatable operation of the CPCS central processing unit (CPU) when processor loading exceeds 70 percent. Provide the basis and analysis for the 70 percent CPU loading criteria and describe how the particular tests and analyses proposed in the application will be conducted to verify predictable and repeatable behavior. Include an inspection, tests, analyses, and acceptance criteria (ITAAC) item to include the necessary analysis and test to ensure predictable and repeatable operation of the CPCS system once software development has been completed and for support of future software maintenance.

10 CFR 50.55a(h)(3) requires compliance with IEEE Std. 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” and the correction sheet dated January 30, 1995. Clause 5.5 of IEEE Std. 603-1991 requires the safety systems shall be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. Clause 5.15, "Reliability," requires, in part, that for those systems for which either quantitative or qualitative reliability goals have been established, appropriate analysis of the design shall be performed in order to confirm that such goals have been achieved. 10 CFR 52.47(b)(1), requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

Technical Report APR1400-A-J-NR-14004-P, Rev. 0, "Common Q Platform Supplemental Information in Support of the APR1400 Design Certification," states, in part, that the maximum load of the Common Q platform CPU (AC160) to be used for the APR1400 CPCS system needs to be raised to 75 percent, which exceeds the 70 percent CPU load limit as specified in the Common Q Topical Report, WCAP-16097-P-A, Rev. 3. In addition, many restrictions for configuration and programming have been proposed in the above technical report APR1400-A-J-NR-14004-P, so the task processing and communication in the CPCS could be deterministic. Particular tests are proposed to be conducted to ensure that the CPCS system will behave in a predictable and repeatable manner. However, the above Common Q Topical Report specifies that the maximum CPU load must not exceed a value of 70 percent to ensure the deterministic communication. Describe how the CPCS system will be able to reliably perform all scheduled CPU tasks when the CPU load exceeds 70 percent. The description should include the basis

for the CPU loading criteria, analysis, and/or outline the analysis to be performed that demonstrates reliable performance for CPU loading once the software is completed. Also, describe how the proposed tests will be conducted to verify the deterministic communication and include an ITAAC to ensure that the necessary analysis and tests will be conducted to ensure the CPCS CPU tasks behave in a predictable and repeatable manner.


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