REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**43-7887**
SRP Section:	**07.01 Instrumentation and Controls - Introduction**
Application Section:	**7.1**
Date of RAI Issue:	**06/22/2015**

Question No. 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.

Revised Response

10 CFR 50.54(jj) and 10 CFR 50.55(i) do not apply to design certification applicants, and 10 CFR 50.55a(a)(1) has been modified to remove quality requirements. Therefore, references to 10 CFR 50.55a(a)(1) will be deleted, and no references to 10 CFR 50.54(jj) and 10 CFR 50.55(i) will be made in DCD Tier 2 sections.

Impact on DCD

DCD Tier 2, Table of Contents, Sections 7.1.2.2, 7.1.5, 7.4.2, 7.4.5, 7.6.2.1, 7.6.4, 7.9.3, 7.9.5, and Table 7.1-1 will be revised, as indicated in Attachment 1 to this response.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

Section 3.1.f of technical report APR1400-Z-J-NR-14001-P/NP, “Safety I&C System,” will be deleted, as indicated in Attachment 2 to this response.

CHAPTER 7 – INSTRUMENTATION AND CONTROLS

TABLE OF CONTENTS

7.1.2.1.4 All Other Systems Required for Safety

The design bases for all other systems required for safety are described in Section 7.6.

7.1.2.1.5 Interlocks

The interlocks for safety instrumentation are described in Subsections 7.2.1.7 and 7.3.1.6 and Section 7.6.

7.1.2.1.6 Bypasses

The bypasses for safety instrumentation are described in Subsections 7.2.1.6 and 7.3.1.5.

7.1.2.1.7 Diversity

The diversity for safety instrumentation is described in Subsections 7.2.1.9, 7.2.2.4, and 7.3.2.4.

~~10 CFR 50.54(jj) and 10 CFR 50.55(i)~~

7.1.2.1.8 Instrumentation Protection

The safety instrumentation protection is described in Chapter 3.

Deleted

7.1.2.2 ~~Conformance with 10 CFR 50.55a(a)(1)~~

~~The I&C 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.~~

7.1.2.3 Conformance with 10 CFR 50.55a(h)(2)

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 has two channels of a reed switch position transmitter (RSPT) for each control element assembly. The alternative to Clause 5.6 of IEEE Std. 603 is described in the Safety I&C System Technical Report.

~~The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.54(jj) and 10 CFR 50.55 (i) (Reference 8). 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 (Reference9), Clause 5.3. Compliance to IEEE Std. 603-1991 is described in Appendix A of the Safety I&C System Technical Report.~~

7.1.5 References

1. APR1400-Z-J-NR-14003-P, “Software Program Manual,” KHNP, November 2014.

2. APR1400-Z-J-NR-14001-P, “Safety I&C System,” KHNP, November 2014.

3. Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Rev. 4, U.S. Nuclear Regulatory Commission, June 2006.

4. NUREG-0737, “Clarification of TMI Action Plan Requirements,” Item II.F.2, “Instrumentation for detection of inadequate core cooling,” U.S. Nuclear Regulatory Commission, November 1980.

5. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs” Item II.T, “Control Room Annunciator (Alarm) Reliability,” U.S. Nuclear Regulatory Commission, April 2, 1993.

6. NUREG-0696, “Functional Criteria for Emergency Response Facilities,” U.S. Nuclear Regulatory Commission, 1981.

7. APR1400-Z-J-NR-14002-P, “Diversity and Defense-in-Depth,” KHNP, November 2014.

Deleted

8. ~~10 CFR 50.55a(a)(1), “Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for Systems Important to Safety,” U.S. Nuclear Regulatory Commission.~~

9. IEEE Std. 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 1991.

10. 10 CFR 50.55a(h)(2), “Codes and Standards, Protection Systems,” U.S. Nuclear Regulatory Commission.

11. 10 CFR 50.55a(h)(3), “Codes and Standards, Safety Systems,” U.S. Nuclear Regulatory Commission.

12. 10 CFR 50.34(f)(2)(v), “Bypass and Inoperable Status Indication,” [I.D.3], U.S. Nuclear Regulatory Commission.

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards," U.S. Nuclear Regulatory Commission.~~

~~10 CFR 50.54(jj) and 10 CFR 50.55(i)~~

Deleted

~~Quality Standards~~

Table 7.1-1 (1 of 6)

Regulatory Requirements Applicability Matrix

Applicable Criteria		Title	I&C System							Section in APR1400 DCD
			RTS	ESF System	QIAS-P	QIAS-N	PCS	P-CCS	DAS	
10 CFR Part 50										
1	~~50.55a(a)(1)~~	~~Quality Standards and Records for Systems Important to Safety~~	~~×~~	~~×~~	~~×~~	~~×~~	~~×~~	~~×~~	~~×~~	~~7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9~~
2	50.55a(h)(2)	Protection Systems	×	×						7.2, 7.3, 7.9
3	50.55a(h)(3)	Safety Systems	×	×	×					7.2, 7.3, 7.5, 7.6, 7.9
4	50.34(f)(2)(v)	Bypass and Inoperable Status Indication	×	×	×	×				7.2, 7.3, 7.5, 7.6, 7.9
5	50.34(f)(2)(xi)	Direct Indication of Relief and Safety Valve Position			×					7.5
6	50.34(f)(2)(xii)	Auxiliary Feedwater System Automatic Initiation and Flow Indication	×	×	×					7.2, 7.3, 7.5
7	50.34(f)(2)(xiv)	Containment Isolation Systems	×	×	×					7.2, 7.3, 7.5
8	50.34(f)(2)(xvii)	Accident Monitoring Instrumentation			×	×				7.5
9	50.34(f)(2)(xviii)	Instrumentation for the Detection of Inadequate Core Cooling			×					7.5
10	50.34(f)(2)(xix)	Instruments for Monitoring Plant Conditions Following Core Damage			×					7.5
11	50.34(f)(2)(xx)	Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves			×					7.4, 7.5
12	50.62	Requirements for Reduction of Risk from Anticipated Transients without Scram							×	7.8
10 CFR Part 50, Appendix A GDC										
13	GDC 1	Quality Standards and Records	×	×	×	×	×	×	×	7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9
14	GDC 2	Design Bases for Protection against Natural Phenomena	×	×	×	×	×	×	×	7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9

7.4.2 Design Basis Information

Safe shutdown design, including the design of the RSR, is based on the following applicable codes and standards:

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards" (Reference 6)~~

a. 10 CFR 50.34(f)(2)(xx) "Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves" [II.G.1] (Reference 5)

Deleted

b. ~~10 CFR 50.55a(a)(1), "Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for Systems Important to Safety" (Reference 6)~~

c. 10 CFR 50.55a(h), "Codes and Standards, Protection and Safety Systems" (Reference 7)

d. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records" (Reference 8)

e. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection against Natural Phenomena" (Reference 9)

f. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effect Design Bases" (Reference 10)

g. 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and Control" (Reference 11)

h. 10 CFR Part 50, Appendix A, GDC 19, "Control Room" (Reference 12)

i. 10 CFR Part 50, Appendix A, GDC 24, "Separation of Protection and Control Systems" (Reference 13)

j. 10 CFR Part 50, Appendix A, GDC 34, "Residual Heat Removal" (Reference 14)

k. 10 CFR Part 50, Appendix A, GDC 35, "Emergency Core Cooling" (Reference 15)

l. 10 CFR Part 50, Appendix A, GDC 38, "Containment Heat Removal" (Reference 16)

7.4.3.3.3 Plant Load Rejection, Turbine Trip, and Loss of Offsite Power

In the event of a LOOP associated with plant load rejection or turbine trip, the power for safe shutdown is provided by the EDGs. The EDGs provide power for operation of pumps and valves; the batteries or EDGs via the battery chargers provide power for operation of instrumentation and control systems required to actuate and control essential components.

7.4.3.3.4 Restrictive Setpoints

There are no restrictive setpoints for the APR1400.

7.4.4 Combined License Information

No combined license (COL) information is required with regard to Section 7.4.

7.4.5 References

1. Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, April 2009.

2. IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 1991.

3. IEEE Std. 7-4.3.2-2003, "IEEE Standard Design for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2003.

4. APR1400-Z-J-NR-14001-P, "Safety I&C System," KHNP, November 2014.

5. 10 CFR 50.34(f)(2)(xx), "Power for Pressurizer Level Indication and Controls for Pressurizer Relief and Block Valves," [II.G.1], U.S. Nuclear Regulatory Commission.

Deleted

6. ~~10 CFR 50.55a(a)(1), "Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for systems Important to Safety," U.S. Nuclear Regulatory Commission.~~

7. 10 CFR 50.55a(h), "Codes and Standards, Protection and Safety Systems," U.S. Nuclear Regulatory Commission.

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards," U.S. Nuclear Regulatory Commission.~~

7.6.2.1 Applicable Codes and Regulations

The interlock systems important to safety are designed to comply with the following codes and regulations:

a. 10 CFR 50.34(f)(2)(v), "Bypass and Inoperable Status Indication" (Reference 2)

The BISI described in Subsection 7.6.1 is designed in accordance with 10 CFR 50.34(f)(2)(v).

The BISI of the interlock systems important to safety is available on the information processing system (IPS) and qualified indication and alarm system - non-safety (QIAS-N).

Deleted

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards" (Reference 4)~~

b. ~~10 CFR 50.55a(a)(1), "Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for Systems Important to Safety" (Reference 4)~~

~~The interlock systems important to safety are defined as safety grade according to ANSI/ANS-51.1 (Reference 3). This is for conformance with IEEE Std. 603, Clause 5.3.~~

~~10 CFR 50.54(jj) and 10 CFR 50.55(i).~~

~~The interlock systems important to safety are tested and inspected to quality standards commensurate with the importance of the safety function to be performed in accordance with 10 CFR 50.55a(a)(1).~~

c. 10 CFR 50.55a(h)(2), "Codes and Standards, Protection Systems" (Reference 5)

The important-to-safety interlock systems described in Subsections 7.6.1.1, 7.6.1.3, and 7.6.1.4 are designed in accordance with 10 CFR 50.55a(h)(2) as follows:

The interlock systems important to safety consist of four independent divisions except the SCS suction line relief valves, which consist of two divisions. The protection division is physically separated and electrically isolated from the other protection divisions. All equipment/components used for safety-related functions are qualified as safety related. The failures of non-safety systems cannot prevent any interlock system important to safety from performing its safety function.

7.6.4 Combined License Information

No combined license (COL) information is required with regard to Section 7.6.

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards," U.S. Nuclear Regulatory Commission.~~

7.6.5 References

1. IEEE Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 1991.

2. 10 CFR 50.34(f)(2)(v), "Bypass and Inoperable Status Indication," [I.D.3], U.S. Nuclear Regulatory Commission.

3. ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," American Nuclear Society, 1983.

Deleted

4. ~~10 CFR 50.55a(a)(1), "Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for Systems Important to Safety," U.S. Nuclear Regulatory Commission.~~

5. 10 CFR 50.55a(h)(2), "Codes and Standards, Protection Systems," U.S. Nuclear Regulatory Commission.

6. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

7. 10 CFR 50.34(f)(2)(xi), "Direct Indication of Relief and Safety Valve Position," [II.D.3], U.S. Nuclear Regulatory Commission.

8. Regulatory Guide 1.75, "Criteria for Independence of Electrical Safety Systems," Rev. 3, U.S. Nuclear Regulatory Commission, February 2005.

9. IEEE Std. 338-1987, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generation Station Safety Systems," Institute of Electrical and Electronics Engineers, 1987.

10. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," Rev. 0, U.S. Nuclear Regulatory Commission, February 1972.

7.9.3 Analysis

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards" (Reference 8).~~

The data communication systems (1) comply with the recommendations in the regulatory guides and industry codes and standards that are applicable to these systems, (2) are in conformance to the requirements of GDC 1 (Reference 13) ~~and 10 CFR 50.55a(a)(1) (Reference 8).~~

A reliability model is created to represent the hardware implementation of the data communication systems. The model is used to determine the estimated reliability and availability of data communication systems. The analysis is based on reliability data provided by equipment manufacturers.

The FMEA demonstrates that failures in data communication systems do not adversely affect the safety function or cause erroneous safety function actuation.

The results of the analysis of the data communication systems are provided in Appendix C of the Safety I&C System Technical Report. These results show compliance with the staff positions in DI&C-ISG-04.

7.9.4 Combined License Information

No combined license (COL) information is required with regard to Section 7.9.

7.9.5 References

1. IEEE Std. 7-4.3.2-2003, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2003.

2. DI&C-ISG-04, "Highly-Integrated Control Rooms – Communications Issues (HICRc)," Rev. 1, U.S. Nuclear Regulatory Commission, March 2009.

3. APR1400-Z-J-NR-14001-P, "Safety I&C System," KHNP, November 2014.

4. APR1400-Z-J-NR-14003-P, "Software Program Manual," KHNP, November 2014.

5. Regulatory Guide 1.75, "Criteria for Independence of Electrical Safety Systems," Rev. 3, U.S. Nuclear Regulatory Commission, February 2005.

6. Regulatory Guide 1.180, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems,” Rev. 1, U.S. Nuclear Regulatory Commission, October 2003.

7. APR1400-Z-J-NR-14002-P, “Diversity and Defense-in-Depth,” KHNP, November 2014.

Deleted

8. ~~10 CFR 50.55a(a)(1), “Domestic Licensing of Production and Utilization Facilities, Codes and Standards, Quality Standards for Systems Important to Safety.” U.S. Nuclear Regulatory Commission~~

9. Regulatory Guide 1.152, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants,” Rev. 3, U.S. Nuclear Regulatory Commission, July 2011.

10. MIL-STD-461E, “Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment.” August, 1999.

11. IEC 61000-4 Series, “Electromagnetic Compatibility-Testing and Measurement Techniques,” International Electrotechnical Commission.

12. NUREG-0800, Standard Review Plan, BTP 7-19, “Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems,” Rev. 6, U.S. Nuclear Regulatory Commission, July 2012.

13. 10 CFR Part 50, Appendix A, General Design Criterion 1, “Quality Standards and Records,” U.S. Nuclear Regulatory Commission.

~~10 CFR 50.54(jj) and 10 CFR 50.55(i), "Quality Standards," U.S. Nuclear Regulatory Commission.~~

3 APPLICABLE CODES AND REGULATIONS

This section describes the compliance of the safety I&C system with the applicable codes and regulations. The system's compliance with IEEE Std. 603-1991, IEEE Std. 7-4.3.2-2003, NRC Interim Staff Guidance (ISG) DI&C-ISG-04, "Highly-Integrated Control Rooms – Communications Issues" (Reference 4), and alternative to independence requirements of IEEE Std. 603-1991 are addressed in Appendices A, B, C, and D of this report, respectively.

3.1 10 CFR Part 50 and 52

a. 10 CFR 50.34(f)(2)(v), "Bypass and Inoperable Status Indication"

The indications of bypasses and inoperable status of the safety I&C system are available on the operator module (OM), maintenance and test panel (MTP), qualified indication and alarm system - non-safety (QIAS-N) and information processing system (IPS) displays.

See compliance with Regulatory Guide (RG) 1.47 in Section 3.4.3.

b. 10 CFR 50.34(f)(2)(xii), "Auxiliary Feedwater System Automatic Initiation and Flow Indication"

The low steam generator (SG) water level trip signal initiates a reactor trip when the measured water level in a SG's downcomer region falls to a low preset value. Separate initiations are provided for the reactor protection system (RPS) and auxiliary feedwater actuation system (AFAS) to allow different setpoints for reactor trips and auxiliary feedwater actuations.

The AFAS continues to deliver auxiliary feedwater to the SG until a preset water level has been reestablished. Manual actuation is provided to permit the operator to actuate the AFAS.

Auxiliary feedwater flow rate is displayed on the QIAS-N, IPS, and diverse indication system (DIS).

c. 10 CFR 50.34(f)(2)(xiv), "Containment Isolation Systems"

The containment isolation actuation system (CIAS) is provided to mitigate the release of radioactive material during an accident by actuating the containment isolation valves (CIVs) which close the process lines penetrating the containment.

d. 10 CFR 50.34(f)(2)(xi), "Direct Indication of Relief and Safety Valve Position"
10 CFR 50.34(f)(2)(xvii), "Instrumentation to Measure, Record and Readout in the Control Room"
10 CFR 50.34(f)(2)(xviii), "Unambiguous Indication of Inadequate Core Cooling"
10 CFR 50.34(f)(2)(xix), "Instrumentation for Monitoring Plant Conditions Following an Accident"
10 CFR 50.34(f)(2)(xx), "Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators"

Types B and C accident monitoring instrumentation are displayed on the QIAS-P, QIAS-N, and IPS. The QIAS-N displays selected variables of Types D and E to support plant safe shutdown and Emergency Operating Procedure (EOP). All variables of Types D and E are displayed on the IPS.

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

The safety I&C system is installed in a mild environment and therefore this criterion is not applicable. This criterion is applicable to instrumentation that interfaces to this system.

Deleted

f. ~~10 CFR 50.55a(a)(1), "Quality Standards"~~

~~10 CFR 50.54(jj) and 10 CFR 50.55(i),~~

~~designed, fabricated, erected, constructed, tested, and inspected to quality standards~~

~~The safety I&C system is defined as Quality Class Q and Safety Class 3 according to ANSI/ANS-51.1-1983 (Reaffirmed 1988, Withdrawn 1998) as described in the Quality Assurance Program Description (QAPD) (Reference 5).~~

g. 10 CFR 50.55a(h), "Protection and Safety Systems"

The safety I&C system is designed to meet the requirements of the requirements of IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. Compliance to IEEE Std. 603-1991 is described in Appendix A of this report.

IEEE Std. 603-1991, Clause 6.7 states, "Capability of a safety system to accomplish its safety functions shall be retained while sense and command features equipment is in maintenance bypass". The Balance of Plant (BOP) ESFAS functions are 1-out-of-2 logic taken twice except the fuel handling area emergency ventilation actuation signal (FHEVAS) initiation signal that performs 1-out-of-2 logic taken once. The detailed compliance with IEEE Std. 603-1991 is described in Appendix A.

The CPCS has two channels of reed switch position transmitter (RSPT) for each control element assembly (CEA). The alternative to Clause 5.6 of IEEE Std. 603-1991 to satisfy the independence requirement is described in Appendix D.

h. 10 CFR 50.62, "Requirements for Reduction of Risk from ATWS"

The diverse protection system (DPS) is designed to satisfy Anticipated Transients Without Scram (ATWS) requirements and is described in the Diversity and Defense-in-Depth TeR. The DPS is diverse from the safety I&C system.

The details of the diversity of the scram system are described in Section 4.8 and the conformance to 10 CFR 50.62 is described in Appendix B of the Diversity and Defense-in-Depth TeR.

i. 10 CFR 52.47(b)(1), "ITAAC for Standard Design Certification"

The Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are described in Section 2.5 of the Design Control Document (DCD).

j. 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Power Plants"

The safety I&C system is designed to meet the requirements of 10 CFR 50 Appendix A as described in Section 3.2.

k. 10 CFR 50 Appendix B. "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants"

The safety I&C system is designed to meet the requirements of 10 CFR 50 Appendix B as described in the QAPD.

l. 10 CFR 52.47(a)(2)(iv), "Release of Radioactive Material"

The CCF coping analysis is performed to meet the guideline values of radiation dose. The results of the offsite radiological consequences obtained from the CCF Coping Analysis TeR meet the acceptance criteria required by 10 CFR 52.47.

3.2 10 CFR Part 50 Appendix A, General Design Criteria

a. GDC 1, "Quality Standards and Records"

The QAPD and Quality Assurance Manual (QAM) (Reference 6) comply with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants ".

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

Korea Electric Power Corporation / Korea Hydro & Nuclear Power Co., LTD

Docket No. 52-046

RAI No.:	**43-7887**
SRP Section:	**07.01 Instrumentation and Controls - Introduction**
Application Section:	**7.1**
Date of RAI Issue:	**06/22/2015**

Question No. 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.

Revised Response

10 CFR 50.54(jj) and 10 CFR 50.55(i) do not apply to design certification applicants, and 10 CFR 50.55a(a)(1) has been modified to remove quality requirements. Therefore, references to 10 CFR 50.55a(a)(1) will be deleted, and no references to 10 CFR 50.54(jj) and 10 CFR 50.55(i) will be made in DCD Tier 2 sections.

Compliance with IEEE Std. 603-1991, Clause 5.3 is described in Appendix A of the Safety I&C System Technical Report.

Impact on DCD

DCD Tier 2, Section 7.1.2.2 will be revised, as shown in Attachment 1 to RAI 43-7887, Question 07.01-10.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on any Technical, Topical, or Environmental Report.