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.

Response

10 CFR 50.55a(a)(1) will be replaced with 10 CFR 50.54(jj) and 10 CFR 50.55(i) in applicable 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 on the attached markup.

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 revised as indicated on the attached markup.

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.

7.1.2.1.8 Instrumentation Protection

The safety instrumentation protection is described in Chapter 3.

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

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.

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.

Table 7.1-1 (1 of 6)

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

Quality Standards

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)

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.

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).

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

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

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.

7.6.5 References

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

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.

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.

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

7.9.3 Analysis

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.

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.

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

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

Safety I&C System 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 ".

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.

Response

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) by complying with the requirements of IEEE Std. 603, Clause 5.3. 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

For changes to Sections 7.1.2.2, please see the attachment associated with the response to RAI 43-7887, Question No. 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.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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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. 07.01-12

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].

Response

DCD Tier 2 Sections 7.1.2.2 through 7.1.2.35 will be modified to state that the applicable I&C systems listed in Table 7.1-1 meet the requirements of NRC regulations in order to meet the requirements of 10 CFR 52.47a(2) and 10 CFR 52.47a(3)(i).

Impact on DCD

The DCD Tier 2, Chapter 7, Table of Contents, and Sections 7.1.2.2 through 7.1.2.35 will be revised as indicated on the attached markup.

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.

CHAPTER 7 – INSTRUMENTATION AND CONTROLS

TABLE OF CONTENTS

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

50.34(f)(2)(v)

50.34(f)(2)(xi)

50.34(f)(2)(xii)

50.34(f)(2)(xiv)

50.34(f)(2)(xvii)

50.34(f)(2)(xviii)

50.34(f)(2)(xix)

50.34(f)(2)(xx)

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.

7.1.2.1.8 Instrumentation Protection

The safety instrumentation protection is described in Chapter 3.

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

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.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.55a(h)(2) (Reference 10) as described in Section 3.1 of the Safety I&C System Technical Report

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.

Appendix D of

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.55a(h)(3) (Reference 11) as described in Section 3.1 of the Safety I&C System Technical Report.

7.1.2.4 Conformance with 10 CFR 50.55a(h)(3)

~~The I&C systems that are applicable to 10 CFR 50.55a(h)(3) (Reference 11), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.55a(h)(3).~~

50.34(f)(2)(v)

7.1.2.5 Conformance with 10 CFR ~~50.34f(2)(v)~~

~~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).~~ Display instrumentation provides accurate, complete, and timely information to safety system status by compliance to Clause 5.8.2 (system status indication) and Clause 5.8.3 (indication of bypasses) of IEEE Std. 603. Conformance with IEEE Std. 603 is described in the Safety I&C System Technical Report. Information regarding bypassed and inoperable status is provided in Subsection 7.5.1.3.

50.34(f)(2)(xi)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xi) (Reference 13),

7.1.2.6 Conformance with 10 CFR ~~50.34f(2)(xi)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xi) (Reference 13), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xi),~~ as described in Subsection 7.5.1.1.

50.34(f)(2)(xii)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xii) (Reference 14) as described in Section 3.1 of the Safety I&C System Technical Report.

7.1.2.7 Conformance with 10 CFR ~~50.34f(2)(xii)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xii) (Reference 14), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xii).~~ The automatic and manual initiation of the auxiliary feedwater system is described in Subsection 7.3.1.9.

50.34(f)(2)(xiv)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xiv) (Reference 15).

7.1.2.8 Conformance with 10 CFR ~~50.34f(2)(xiv)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xiv) (Reference 15), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xiv).~~ The containment isolation function, including reset of the function, is described in Subsection 7.3.1.9.

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(v) (Reference 12) as described in Section 3.1 of the Safety I&C System Technical Report.

50.34(f)(2)(xvii)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xvii) (Reference 16),

7.1.2.9 Conformance with 10 CFR ~~50.34f(2)(xvii)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xvii) (Reference 16), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xvii),~~ as described in Subsection 7.5.1.

50.34(f)(2)(xviii)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xviii) (Reference 17), as described in Subsection 7.5.1.2.

7.1.2.10 Conformance with 10 CFR ~~50.34f(2)(xviii)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xviii) (Reference 17), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xviii), as described in Subsection 7.5.1.1.~~

50.34(f)(2)(xix)

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xix) (Reference 18), as described in Subsection 7.5.1.1.

7.1.2.11 Conformance with 10 CFR ~~50.34f(2)(xix)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xix) (Reference 18), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xix).~~

50.34(f)(2)(xx)

7.1.2.12 Conformance with 10 CFR ~~50.34f(2)(xx)~~

~~The I&C systems that are applicable to 10 CFR 50.34f(2)(xx) (Reference 19), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.34(f)(2)(xx).~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.34(f)(2)(xx) (Reference 19), as described in Subsection 7.5.2.1.

7.1.2.13 Conformance with 10 CFR 50.62

~~The I&C systems that are applicable to 10 CFR 50.62 (Reference 20), as shown in Table 7.1-1, are designed in accordance with 10 CFR 50.62, which states in part, "Each pressurized water reactor manufactured by Combustion Engineering must have a diverse scram system from the sensor output to interruption of power to the control rods."~~ The conformance with 10 CFR 50.62 is described in the Diversity and Defense-in-Depth Technical Report.

The applicable I&C systems listed in Table 7.1-1 meet the requirements of 10 CFR 50.62 (Reference 20).

7.1.2.14 Conformance with GDC 1

~~The I&C systems that are applicable to GDC 1 (Reference 21), as shown in Table 7.1-1, are designed in accordance with GDC 1 through compliance with IEEE Std. 603, Clause 5.3.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 1 (Reference 21) as described in Section 3.2 of the Safety I&C System Technical Report.

The quality assurance program description (QAPD) complies with the requirements of 10 CFR Part 50, Appendix B (Reference 22).

7.1.2.15 Conformance with GDC 2

~~The I&C systems that are applicable to GDC 2, as shown in Table 7.1-1, are designed in accordance with GDC 2 through compliance with IEEE Std. 603, Clause 5.4.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 2 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.16 Conformance with GDC 4

~~The I&C systems that are applicable to GDC 4, as shown in Table 7.1-1, are designed in accordance with GDC 4 through compliance with IEEE Std. 603, Clause 5.4.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 4 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.17 Conformance with GDC 10

~~The I&C systems that are applicable to GDC 10, as shown in Table 7.1-1, are designed in accordance with GDC 10.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 10 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.18 Conformance with GDC 13

~~The I&C systems that are applicable to GDC 13, as shown in Table 7.1-1, are designed in accordance with GDC 13.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 13 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.19 Conformance with GDC 15

~~The I&C systems that are applicable to GDC 15, as shown in Table 7.1-1, are designed in accordance with GDC 15.~~

The applicable I&C systems listed in Table 7.1-1 meets the requirement of GDC 15 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.20 Conformance with GDC 16

~~The I&C systems that are applicable to GDC 16, as shown in Table 7.1-1, are designed in accordance with GDC 16.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 16 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.21 Conformance with GDC 19

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 19 as described in Section 3.2 of the Safety I&C System Technical Report.

~~The I&C systems that are applicable to GDC 19, as shown in Table 7.1-1, are designed in accordance with GDC 19.~~ The capabilities with regard to the safe operation of the plant from the MCR during normal and accident conditions are described in Section 7.4.

7.1.2.22 Conformance with GDC 20

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 20.

~~The I&C systems that are applicable to GDC 20, as shown in Table 7.1-1, are designed in accordance with GDC 20.~~ The protection function is described in Sections 7.2 and 7.3.

7.1.2.23 Conformance with GDC 21

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 21 as described in Section 3.2 of the Safety I&C System Technical Report.

~~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.

7.1.2.24 Conformance with GDC 22

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 22 as described in Subsections 7.2.2.3 and 7.3.2.3 as well as in Section 4.1 of Safety I&C System Technical Report.

~~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 position inputs described in Subsection 7.1.2.3.

7.1.2.25 Conformance with GDC 23

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 23.

~~The I&C systems that are applicable to GDC 23, as shown in Table 7.1-1, are designed in accordance with GDC 23.~~ Failure modes and effects analysis (FMEA) for protection systems is described in Subsections 7.2.3.1 and 7.3.3.1.

7.1.2.26 Conformance with GDC 24

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 24 as described in A.5.6 of the Safety I&C System Technical Report.

~~The I&C systems that are applicable to GDC 24, as shown in Table 7.1-1, are designed in accordance with GDC 24.~~ Electrical isolation, physical separation, and communication

independence are maintained between redundant safety divisions and between the safety system and non-safety system.

7.1.2.27 Conformance with GDC 25

~~The I&C systems that are applicable to GDC 25, as shown in Table 7.1-1, are designed in accordance with GDC 25.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 25 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.28 Conformance with GDC 28

~~The I&C systems that are applicable to GDC 28, as shown in Table 7.1-1, are designed in accordance with GDC 28.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 28 as described in Subsections 7.6.2.1 and 7.7.1.1.

7.1.2.29 Conformance with GDC 29

~~The I&C systems that are applicable to GDC 29, as shown in Table 7.1-1, are designed in accordance with GDC 29.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 29 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.30 Conformance with GDC 33

~~The I&C systems that are applicable to GDC 33, as shown in Table 7.1-1, are designed in accordance with GDC 33.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 33 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.31 Conformance with GDC 34

~~The I&C systems that are applicable to GDC 34, as shown in Table 7.1-1, are designed in accordance with GDC 34.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 34 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.32 Conformance with GDC 35

~~The I&C systems that are applicable to GDC 35, as shown in Table 7.1-1, are designed in accordance with GDC 35.~~

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 35 as described in Section 3.2 of the Safety I&C System Technical Report.

7.1.2.33 Conformance with GDC 38

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 38 as described in Section 3.2 of the Safety I&C System Technical Report.

~~The I&C systems that are applicable to GDC 38, as shown in Table 7.1-1, are designed in accordance with GDC 38.~~

7.1.2.34 Conformance with GDC 41

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 41 as described in Section 3.2 of the Safety I&C System Technical Report.

~~The I&C systems that are applicable to GDC 41, as shown in Table 7.1-1, are designed in accordance with GDC 41.~~

7.1.2.35 Conformance with GDC 44

The applicable I&C systems listed in Table 7.1-1 meet the requirements of GDC 44 as described in Section 3.2 of the Safety I&C System Technical Report.

~~The I&C systems that are applicable to GDC 44, as shown in Table 7.1-1, are designed in accordance with GDC 44.~~

7.1.2.36 Conformance with SECY-93-087, Item II.Q

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). The analyses and design features address postulated safety system CCFs and are described in the Diversity and Defense-in-Depth Technical Report.

7.1.2.37 Conformance with SECY-93-087, Item II.T

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). The APR1400 design complies with the requirements as follows:

a. Redundancy

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety.

Major equipment of the IPS such as the computational server, alarm server, historical data storage and retrieval (HDSR) server, and data communication are configured to primary and standby processors.

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. 07.01-13

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.

Response

DCD Tier 2 Section 7.1.2, its subsections, Section 11.5.2.1, and the Diversity and Defense-in-Depth Technical Report will be modified to describe references to applicable APR1400 DCD sections and technical reports that contain information on how regulations are met. For changes to Sections 7.1.2.2 through 7.1.2.35, please see the attachment associated with the response to RAI 43-7887, Question No. 07.01-12. The changes to Tier 2, DCD Table 7.1-1, Section 11.5.2.1, and Appendix B of the Diversity and Defense-in-Depth Technical Report are shown in the attachment associated with this response.

Impact on DCD

DCD Tier 2, Table 7.1-1, Section 11.5.2.1, and Appendix B of the Diversity and Defense-in-Depth Technical Report will be revised as indicated on the attached markup.

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

Technical Report APR1400-Z-J-NR-14002-P/NP, APPENDIX B. will be revised as indicated in the attached markup.

c. Data communication network – information (DCN-I) network for non-safety systems

7.1.1.10 Auxiliary Support Features

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.

7.1.2 Identification of Safety Criteria

Subsections 7.1.2.2 through 7.1.2.75 and Sections 7.2 through 7.6 contain comparisons of the design with the applicable NRC regulatory guides and a description of the degree of compliance with the appropriate design bases, the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A (Reference 21), standards, and other documents used in the design of the systems listed in Subsection 7.1.1.

7.1.2.1 Design Bases

The design bases for each safety I&C system are presented in the relevant sections of this chapter.

7.1.2.1.1 Systems Required for Plant Protection

The design bases for plant protection systems are described in Sections 7.2 and 7.3.

7.1.2.1.2 Systems Required for Safe Shutdown

The design bases for the systems required for safe shutdown are described in Section 7.4.

7.1.2.1.3 Information Systems Important to Safety

The design bases for information systems important to safety are described in Section 7.5.

Compliance with 10 CFR Part 50 and 52 is described in Section 3.1 of the Safety I&C System Technical Report. Compliance with 10 CFR Part 50 Appendix A, General Design Criteria is described in Section 3.2 of the Safety I&C System Technical Report. Compliance with IEEE 603-1991 is described in Appendix A of the Safety I&C System Technical Report.

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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

11.5.2 System Description

11.5.2.1 Monitor Design and Configuration

Process, effluent, and airborne radiation monitors typically consist of components such as a microprocessor, one or more detectors, a shielded detection chamber, a sample pump, flow instrumentation, and associated tubing and cabling.

Each process, effluent, and airborne radiation monitor is located in an easily accessible area and has sufficient shielding to provide reasonable assurance that the required sensitivity is achieved at the design background radiation level for the area. This approach is consistent with NRC RG 8.8 (Reference 28) and NRC RG 8.10 (Reference 29). Instrumentation and sensors are provided to detect component failures such as loss of power, loss of sample flow, check source response failure, and loss of detector signal.

Radiation level signals, alarms, and operation status alarms are generated by each monitor microprocessor and are transmitted to IPS, QIAS, and other interfacing systems. Alarm relay contacts are provided for alert-radiation, high-radiation, and operation status alarms.

For some monitors, the high-radiation alarm contacts are used to initiate control functions to terminate batch releases or to divert flow from one location to another. The operation status alarm is initiated by the microprocessor if conditions indicate that the monitor is not operating properly.

Radiation monitoring equipment is designed for service based on expected environmental conditions during normal operation and AOOs. These conditions include temperature, pressure, humidity, chemical spray (where applicable), and radiation exposure. Post-accident radiation monitors conform with NRC RG 1.97 including equipment qualification, redundancy, power source, channel availability, quality assurance, display and recording, range, interfaces, testing, calibration, and human factors engineering recommendations. Further description of conformance with NRC RG 1.97 is contained in Subsections ~~7.1.2.44~~ and 7.5.2.1.

7.1.2.43

The RMS has an integral activated check source similar to the sample isotope to be detected to monitor proper system response automatically.

APPENDIX B. CONFORMANCE TO 10 CFR 50.62

extracted requirements from

The DPS provides the ATWS mitigation functions required by 10 CFR 50.62. This appendix describes the conformance of the DPS to the requirements of 10 CFR 50.62 (Reference 5). Italic text in this appendix indicates the ~~original requirements of~~ 10 CFR 50.62.

(1) "Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system."

The DPS provides automatic turbine trip and AFWS actuation. Figure 4-2 shows the simplified architecture for diverse automatic AFWS actuation and for the diverse automatic turbine trip. The DPS AFWS actuation is automatically actuated on low steam generator water level in either steam generator.

The DPS turbine trip is also automatically initiated whenever the DPS reactor trip has been actuated. The DPS turbine trip signal will be generated after the initiation of DPS reactor trip signal with three seconds of time delay.

The common safety PLC based platform is used for the reactor trip in the PPS. The DPS is implemented on a FLC platform which is diverse form the common safety PLC platform.

The DPS is designed to perform its function in a fault-tolerant manner, and it is independent from sensor outputs to the shunt trip relays of the RTSS.

Deleted

(2) "Each pressurized water reactor ~~manufactured by Combustion engineering or by Babcock and Wilcox~~ must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods)."

The reactor trip function from the PPS is diverse and independent from the reactor trip function provided by the DPS. The simplified architecture between the reactor trip function from the PPS and diverse reactor trip function from the DPS is shown in Figure 4-2. A DPS reactor trip function is automatically actuated by high pressurizer pressure, high containment pressure, or turbine trip (only if the RPCS is out of service).

a. The common safety platform is used for the reactor trip in the PPS. The diverse reactor trip is provided by the DPS implemented on a diverse FLC platform.

b. The reactor trip from the PPS breaks the power of the CEDM using the undervoltage trip coils of the reactor trip circuit breakers. The diverse reactor trip from the DPS breaks the power of the CEDM using the shunt trip coils of the reactor trip circuit breakers.

c. The RTSS-1 and RTSS-2 reactor trip breakers are diverse each other to ensure that a diverse means exists to break power to the CEDMs.

d. The process instrumentation (PI) sensors for the safety I&C systems are shared by the DPS. The PI sensor signals are electrically isolated in the APC-S prior to being hardwired to the DPS.

(3) "To develop QA guidance for non-safety-related ATWS equipment, the NRC staff both

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: **7.1 - Instrumentation and Controls**

Application Section: **7.1.2.23**

Date of RAI Issue: **06/22/2015**

Question No. 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.

Response

GDC 21 states the redundancy and independence of 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.

Clause 5.1of IEEE Std. 603 corresponds to the requirement that no single failure results in loss of the protection function, as stated in GDC 21. The protection system is designed to comply with Clause 5.1 of IEEE Std. 603. Compliance to IEEE Std. 603-1991, Clause 5.1 is described in Section A.5.1 of the Safety I&C System Technical Report.

The protection system is designed to meet the requirement that removal from service of any component or channel shall not result in loss of the required minimum redundancy. The compliance is described in Sections 7.2.1.8 and 7.3.1.7 of DCD Tier 2, and Section A.5.1 of the Safety I&C System Technical Report.

Section 7.1.2.23 of DCD Tier 2 will be revised as follows:

Current description: 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.

To be revised as follows: The applicable I&C systems listed in Table 7.1-1 are designed to meet the requirement of GDC 21 (Reference 21). Clause 5.1of IEEE Std. 603 corresponds to the requirement that no single failure results in loss of the protection function as stated in GDC 21. The protection system is designed to conform to Clause 5.1 of IEEE Std. 603. Conformance to IEEE Std. 603-1991, Clause 5.1 is described in Section A.5.1 of the Safety I&C System Technical Report. The protection system is designed to meet the requirement that removal from service of any component or channel shall not result in loss of the required minimum redundancy; Sections 7.2.1.8 and 7.3.1.7 describe the safety systems' compliance.

Impact on DCD

Section 7.1.2.23 of DCD Tier 2 will be revised as indicated on the attached mark-up.

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 Reports.

7.1.2.21 Conformance with GDC 19

The I&C systems that are applicable to GDC 19, as shown in Table 7.1-1, are designed in accordance with GDC 19. The capabilities with regard to the safe operation of the plant from the MCR during normal and accident conditions are described in Section 7.4.

7.1.2.22 Conformance with GDC 20

The I&C systems that are applicable to GDC 20, as shown in Table 7.1-1, are designed in accordance with GDC 20. The protection function is described in Sections 7.2 and 7.3.

7.1.2.23 Conformance with GDC 21

~~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.~~

7.1.2.24 Conformance with GDC 22

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 position inputs described in Subsection 7.1.2.3.

7.1.2.25 Conformance with GDC 23

The I&C systems that are applicable to GDC 23, as shown in Table 7.1-1, are designed in accordance with GDC 23. Failure modes and effects analysis (FMEA) for protection systems is described in Subsections 7.2.3.1 and 7.3.3.1.

The applicable I&C systems listed in Table 7.1−1 are designed to meet the requirement of GDC 21 (Reference 21). Clause 5.1of IEEE Std. 603 corresponds to the requirement that no single failure results in loss of the protection function as stated in GDC 21. The protection system is designed to conform to Clause 5.1 of IEEE Std. 603. Conformance to IEEE Std. 603−1991, Clause 5.1 is described in Section A.5.1 of the Safety I&C System Technical Report. The protection system is designed to meet the requirement that removal from service of any component or channel shall not result in loss of the required minimum redundancy; Sections 7.2.1.8 and 7.3.1.7 describe the safety systems' compliance.

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**

Review Section : **07.01 – Instrumentation and Control - Introduction**

Application Section : **Section 7.1**

Date of RAI Issue : **06/22/2015**

Question No. 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.

Response

The diverse actuation system is designed to meet the staff requirements memorandum (SRM) on SECY-93-087, Item II.Q, as described in DCD Tier 2, Section 7.8, and CCF Coping Analysis Technical Report (APR1400-Z-A-NR-14019). To clarify the reference requirements as described in Reference 16 (SRM on SECY-93-087) of NUREG-0800 BTP 7-19, all related DCD and TeR documents will be revised as follows:

1. DCD Tier 2, Section 7.1.2.36:

Current description: Conformance with SECY-93-087, Item II.Q

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).

To be revised as follows: Conformance with SRM on SECY-93-087, Item II.Q

Analyses and design features for diversity and defense-in-depth for the instrumentation and control systems are provided in accordance with the staff requirements memorandum (SRM) on SECY-93-087, Item II.Q (Reference 23), as referenced by NUREG-0800, BTP 7-19 (Reference 73).

2. DCD Tier 2, Section 7.1.5:

Current description: 23. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs” Item II.Q, “Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems,” U.S. Nuclear Regulatory Commission , April 2, 1993.

To be revised as follows: 23. SRM on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs” Item II.Q, “Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems,” U.S. Nuclear Regulatory Commission, July 21, 1993.

3. DCD Tier 2, Sections 1.2.16(4), 1.5.4, 1.5.5(4),7.1.2.36, 7.3.2.4, 7.3.5(9), 7.8, 7.8.2.2, 7.8.5(2), and 18.7.2.1(g)(5), and Table 7.1-1(35):

Current description: SECY-93-087

To be revised as follows: SRM on SECY-93-087

4. Section 3.3.1 of APR1400-Z-J-NR-14002-P, “Diversity and Defense in Depth”

Current description: SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” 1993, Item II.Q, “Defense against Common-Mode Failures in Digital Instrumentation and Control Systems”, and the associated Staff Requirements Memorandum (SRM), 1993 (Reference 7).

To be revised as follows: SRM on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” Item II.Q, “Defense against Common-Mode Failures in Digital Instrumentation and Control Systems”, July 1993 (Reference 7).

5. Reference 7 of APR1400-Z-J-NR-14002-P, “Diversity and Defense in Depth”

Current description: SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs”, April 1993, and the associated Staff Requirements Memorandum, July 1993

To be revised as follows: SRM on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” Item II.Q, “Defense against Common-Mode Failures in Digital Instrumentation and Control Systems”, July 1993

6. Section 3.3.1 of APR1400-Z-J-NR-14001-P, “Safety I&C System”

Current description: SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors”, II.Q, “Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems”

To be revised as follows: SRM on SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors”, Item II.Q, “Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems”

Impact on DCD

DCD Tier 2, Section 7.1.36 will be revised as indicated on the Attachment 1.

DCD Tier 2, Section 7.1.5 will be revised as indicated on the Attachment 2.

DCD Tier 2, Sections 1.2.16(4), 1.5.4, 1.5.5(4), 7.1.2.36, 7.3.2.4, 7.3.5(9), 7.8, 7.8.2.2, 7.8.5(2), and 18.7.2.1(g)(5), and Table 7.1-1(35) will be revised as indicated on the Attachment 3.

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

Diversity and Defense-in-Depth TeR(APR1400-Z-J-NR-14002-P/NP) will be revised as indicated on the Attachments 4 and 5.

Safety I&C System TeR(APR1400-Z-J-NR-14001-P/NP) will be revised as indicated on the Attachment 6.

7.1.2.33 Conformance with GDC 38

The I&C systems that are applicable to GDC 38, as shown in Table 7.1-1, are designed in accordance with GDC 38.

7.1.2.34 Conformance with GDC 41

The I&C systems that are applicable to GDC 41, as shown in Table 7.1-1, are designed in accordance with GDC 41.

7.1.2.35 Conformance with GDC 44

The I&C systems that are applicable to GDC 44, as shown in Table 7.1-1, are designed in accordance with GDC 44.

Conformance with SRM on SECY-93-087, Item II.Q

7.1.2.36 ~~Conformance with SECY-93-087, Item II.Q~~

~~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).~~ The analyses and design features address postulated safety system CCFs and are described in the Diversity and Defense-in-Depth Technical Report.

7.1.2 Analyses and design features for diversity and defense-in-depth for the instrumentation and control systems are provided in accordance with the staff requirements memorandum (SRM) on SECY-93-087, Item II.Q (Reference 23), as referenced by NUREG-0800, BTP 7-19 (Reference 73).

The system requirements in accordance with SECY-93-087, Item II.I (Reference 5). The APR1400 design complies with the requirements as follows:

a. Redundancy

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety.

Major equipment of the IPS such as the computational server, alarm server, historical data storage and retrieval (HDSR) server, and data communication are configured to primary and standby processors.

13. 10 CFR 50.34(f)(2)(xi), "Direct Indication of Relief and Safety Valve Position," [II.D.3], U.S. Nuclear Regulatory Commission.

14. 10 CFR 50.34(f)(2)(xii), "Auxiliary Feedwater System Automatic Initiation and Flow Indication," [II.E.1.2] U.S. Nuclear Regulatory Commission.

15. 10 CFR 50.34(f)(2)(xiv), "Containment Isolation Systems," [II.E.4.2], U.S. Nuclear Regulatory Commission.

16. 10 CFR 50.34(f)(2)(xvii), "Accident Monitoring Instrumentation," [II.F.1], U.S. Nuclear Regulatory Commission

17. 10 CFR 50.34(f)(2)(xviii), "Instrumentation for Detection of Inadequate Core Cooling," [II.F.2], U.S. Nuclear Regulatory Commission.

18. 10 CFR 50.34(f)(2)(xix), "Instruments for Monitoring Plant Conditions Following Core Damage," [II.F.3], U.S. Nuclear Regulatory Commission.

19. 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.

20. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

21. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

SRM on SECY-93-087

22. 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," U.S. Nuclear Regulatory Commission.

23. ~~SECY-93-087~~," Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs" Item II.Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission , ~~April 2, 1993.~~ July 21, 1993.

24. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," U.S. Nuclear Regulatory Commission, various dates and revisions.

Two 50 percent capacity condensate storage tanks store and supply the condensate, as a readily available source of deaerated condensate for makeup, to the condenser. The condensate storage tank is described further in Subsection 9.2.6.

1.2.15 Combined License Information

COL 1.2(1) The COL applicant is to prepare a complete and detailed site plan.

1.2.16 References

1. 10 CFR 50.34, “Contents of Applications; Technical Information,” U.S. Nuclear Regulatory Commission.

2. 10 CFR Part 50, Appendix A, General Design Criterion 54, “Systems Penetrating Containment,” U.S. Nuclear Regulatory Commission.

3. 10 CFR 50.62, “Requirements for the Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants,” U.S. Nuclear Regulatory Commission.

4. Staff Requirements Memorandum ~~to~~ on SECY-93-087, ~~II.Q~~ Item II.Q, “Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems,” ” U.S. Nuclear Regulatory Commission, 1993

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

6. 10 CFR Part 50, Appendix A, General Design Criterion 60, “Control of Release of Radioactive Materials to the Environment,” U.S. Nuclear Regulatory Commission.

7. 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” U.S. Nuclear Regulatory Commission.

The safety I&C system consists of the plant protection system (PPS), core protection calculator system (CPCS), engineered safety features – component control system (ESF-CCS), and the qualified indication and alarm system – P (QIAS-P). The control and monitoring system includes the power control system (PCS), process-component control system, qualified indication and alarm system – non-safety (QIAS-N), and information processing system (IPS). The DAS is composed of the diverse protection system (DPS), diverse indication system (DIS), and diverse manual ESF actuation (DMA) switch. The HSI system includes the compact workstation-based operator console with an information flat panel display and ESF-CCS soft control module (ESCM), large display panel, safety console with ESCM / manual switches / operator module / display device in the main control room, compact workstation-based operator console with ESCM, and a shutdown overview panel in the remote shutdown room.

The safety I&C system is implemented on the four channels of common programmable logic controller qualified for Class 1E grade in accordance with IEEE Std. 603 (Reference 1) and IEEE Std. 7-4.3.2 (Reference 2), and each channel is located in the separate I&C equipment room.

The software for the digital I&C system is designed, verified, and validated in accordance with software life-cycle process conforming with NRC RG 1.152 (Reference 3).

The control and monitoring system is implemented on a distributed control system.

The diversity and defense-in-depth analysis is performed to demonstrate that the DAS and control system meet ~~SECY-93-087, II.Q~~ (Reference 4) in case of software common-cause failure in the safety I&C system. The DAS is implemented on the platform diverse from the safety I&C system and control system.

the SRM on SECY-93-087, Item II.Q

The data communication system provides a high-speed and error-free communication path between each system and within a system.

The HSI system is designed in accordance with the human factors engineering (HFE) program to provide reasonable assurance that the HFE design is properly developed and effectively implemented. The HFE program objectives for the NPP design are that the design is human-centered, it incorporates HFE principals and methods, and is developed

according to a systematic top-down approach. In accordance with applicable requirements of the HFE process elements, the HFE program plan provides reasonable assurance that the HSI design effectively supports the operator and allows consequential operator errors to be minimized. The HFE program is in effect at least from the start of the design cycle through completion of initial plant startup test program to conform with NUREG-0711 (Reference 5).

1.5.5 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. 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

3. 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.

4. ~~SECY-93-087, II.Q~~, "Defense against Common-Mode Failures in Digital Instrumentation and Control Systems, " U.S. Nuclear Regulatory Commission, July 1993.

SRM on SECY-93-087, Item II.Q

5. NUREG-0711, "Human Factors Engineering Program Review Model," Rev.3, U.S. Nuclear Regulatory Commission, November 2012.

7.1.2.33 Conformance with GDC 38

The I&C systems that are applicable to GDC 38, as shown in Table 7.1-1, are designed in accordance with GDC 38.

7.1.2.34 Conformance with GDC 41

The I&C systems that are applicable to GDC 41, as shown in Table 7.1-1, are designed in accordance with GDC 41.

7.1.2.35 Conformance with GDC 44

The I&C systems that are applicable to GDC 44, as shown in Table 7.1-1, are designed in accordance with GDC 44.

SRM on

7.1.2.36 Conformance with SECY-93-087, Item II.Q

SRM on

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). The analyses and design features address postulated safety system CCFs and are described in the Diversity and Defense-in-Depth Technical Report.

7.1.2.37 Conformance with SECY-93-087, Item II.T

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). The APR1400 design complies with the requirements as follows:

a. Redundancy

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety.

Major equipment of the IPS such as the computational server, alarm server, historical data storage and retrieval (HDSR) server, and data communication are configured to primary and standby processors.

APR1400 DCD TIER 2

Table 7.1-1 (3 of 6)

SRM on SECY-93-087, Item II.Q

Applicable Criteria		Title	I&C System							Section in APR1400 DCD
			RTS	ESF System	QIAS-P	QIAS-N	PCS	P-CCS	DAS	
Staff Requirements Memoranda										
35	~~SECY-93-087, Item II.Q~~	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	×	×					×	7.2, 7.3, 7.8, 7.9
36	SECY-93-087, Item II.T	Control Room Annunciator (Alarm) Reliability				×				7.5, 7.9
NRC Regulatory Guides										
37	NRC RG 1.22	Periodic Testing of Protection System Actuation Functions	×	×						7.2, 7.3,, 7.9
38	NRC RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	×	×		×				7.2, 7.3, 7.5, 7.6, 7.9
39	NRC RG 1.53	Application of the Single-Failure Criterion to Safety Systems	×	×	×					7.2, 7.3, 7.4, 7.5, 7.6, 7.9
40	NRC RG 1.62	Manual Initiation of Protective Actions	×	×					×	7.2, 7.3, 7.8
41	NRC RG 1.75	Criteria for Independence of Electrical Safety Systems	×	×	×	×	×	×	×	7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9
42	NRC RG 1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants			×	×				7.5
43	NRC RG 1.105	Setpoints for Safety-Related Instrumentation	×	×	×	×				7.2, 7.3, 7.4, 7.5, 7.6, 7.9
44	NRC RG 1.118	Periodic Testing of Electric Power and Protection Systems	×	×	×	×				7.2, 7.3, 7.4, 7.5, 7.6, 7.9
45	NRC RG 1.151	Instrument Sensing Lines	×	×	×					7.2, 7.3, 7.5,
46	NRC RG 1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants	×	×	×					7.2, 7.3, 7.5, 7.9

switch are input to the CIM, and the CIM prioritizes the control signals according to the priority logic as described in the Component Interface Module Technical Report.

SRM on

According to BTP 7-19 (Reference 8) and SECY-93-087, Item II.Q, Position 4 (Reference 9), the event of a postulated CCF of both the PPS/ESF-CCS and a LOOP is evaluated.

Under the LOOP condition, the EDG is started to supply power into the safety buses. However, if a disabled condition is initiated by software CCF of the PPS and ESF-CCS, necessary power buses are supplied by the alternate alternating current gas turbine generator (AAC GTG) through manual action.

The EDG start/stop function can be accomplished through the manual operation of the local switches for the applicable breakers. The load shedding and the loading sequencer can be carried by the manual operation of the local switches for the applicable load.

The ESF system provides an echelon of defense, as described in the Diversity and Defense-in-Depth Technical Report (Reference 10).

7.3.2.5 System Testing and Inoperable Surveillance

The ESF system integrity is confirmed through periodic testing during power operation or shutdown. The tests cover the trip actions from sensor input to actuation device. The system test does not interfere with the protective function. The tests comply with the criteria of IEEE Std. 338 (Reference 11), which are endorsed by NRC RG 1.118 (Reference 12) and NRC RG 1.22 (Reference 13). The test intervals are specified in Chapter 16, Technical Specifications. The simplified test logic diagram for the ESF-CCS is shown in Figure 7.3-22.

The test equipment consists of divisionalized MTP, ITP, and the associated interface circuits. Test results are verified at the MTP.

Bypasses and the inoperable status of the safety system are displayed at the MTP and OM in accordance with NRC RG 1.47 (Reference 14).

Status information including input variable value, setpoint, trip, pre-trip, initiation, trip channel bypass, and operating bypass is displayed at the MTP, OM and IPS.

8. 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.

9. SECY-93-087, Item II.Q, ~~Position 4,~~ "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission, ~~April 2~~, 1993.

10. APR1400-Z-J-NR-14002-P, "Diversity and Defense-in-Depth," KHNP, November 2014.

11. 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.

12. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," Rev. 3, U.S. Nuclear Regulatory Commission, April 1995.

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

14. Regulatory Guide 1.47 "Bypassed and Inoperable Status indication for Nuclear Power Plant Safety Systems," Rev. 1, U.S. Nuclear Regulatory Commission, February 2010.

15. APR1400-Z-J-NR-14004-P, "Uncertainty Methodology and Application for Instrumentation," KHNP, November 2014.

16. APR1400-Z-J-NR-14005-P, "Setpoint Methodology for Plant Protection System," KHNP, November 2014.

17. ANSI/ISA-S67.04-1994, "Setpoints for Nuclear Safety-Related Instrumentation," International Society of Automation, 1994.

18. Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Rev. 3, U.S. Nuclear Regulatory Commission, December 1999.

19. IEEE Std. 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2003.

SRM on

July 21

7.8 Diverse Instrumentation and Control Systems

The diverse actuation system (DAS) consists of the diverse instrumentation and control (I&C) systems that are provided to protect against potential common-cause failure (CCF) of digital safety I&C systems including the plant protection system (PPS) and engineered safety features-component control system (ESF-CCS). The design has sufficient diversity and defense-in-depth to tolerate the following beyond design basis events:

a. Anticipated transients without scram (ATWS), which is defined as an anticipated operational occurrence (AOO) followed by failure of the reactor trip portion of the PPS.

b. An AOO or a postulated accident (PA) concurrent with a software CCF that prevents the safety I&C systems from performing their required functions.

The DAS consists of the diverse protection system (DPS), the diverse manual engineered safety features (ESF) actuation (DMA) switches, and the diverse indication system (DIS).

SRM on SECY-93-087

For the ATWS mitigation, the DPS is provided to meet the requirements of 10 CFR 50.62 (Reference 1). In addition, the DPS, DIS, and DMA switches are provided to comply with ~~SECY-93-087~~ (Reference 2) and BTP 7-19 (Reference 5). The DPS and DMA switches are independent and diverse from the PPS and ESF-CCS. The DMA switches are located in the main control room (MCR) for manual ESF actuation of critical safety functions.

A reactor trip, turbine trip, auxiliary feedwater actuation, and safety injection actuation functions are included in the DPS. These functions are provided to assist the mitigation of the ATWS and to mitigate the effects of a postulated CCF within the PPS and ESF-CCS. The DMA switches are provided to permit the operator to actuate ESF systems in a timely manner from the MCR after a postulated CCF of the PPS and ESF-CCS. In addition, the DIS provides diverse indications to monitor critical variables and control the heater power for proper HJTC output signal level, when the CCF of digital I&C safety systems occurs.

The DPS and DMA switches are connected to the component interface module (CIM) to cope with a CCF of the PPS and ESF-CCS. The interface description of the DPS and DMA switches are described in the Component Interface Module Technical Report (Reference 12).

System Testing

A channel functional test is performed for the DMA switches by manual actuation of each function. This testing is performed during plant outages to verify that the actuation switch can actuate the components.

Environmental Qualification

The DMA switches are qualified to perform their intended protective function during design basis events.

Independence from the Protection Systems

The DMA switches are connected directly to fan-out devices in the MCR safety console to distribute the manual ESF actuation signals to individual component controls. The signals are hardwired to the CIM in the ESF-CCS loop controller cabinet through isolation devices. The DMA switches compose a non-safety system, but they are designed with Class 1E hardware and are energized using Class 1E power. Therefore, isolation devices are provided for electrical isolation between the DMA switches and the CIM.

Single Failure

Because the DMA switches are a non-safety system, the DMA switches do not need to meet the single failure criterion for actuation.

Diversity

The DMA switches are diverse from the manual and automatic logic functions performed by digital equipment in the PPS and ESF-CCS. The DMA switches are connected to priority logic of the CIM, and priority logic is implemented by hardware devices.

Diversity and Defense-in-Depth

SRM on

The DMA switches are designed to comply with the regulatory position in SECY-93-087, Item II.Q, and with BTP 7-19. The DMA switches provide manual control capability that is used in the event of a postulated software CCF of the digital computer logic within the PPS and ESF-CCS.

The qualitative evaluation assumes that the automatic actuations of safety functions in the PPS and ESF-CCS and the capability for manual actuation using these systems are precluded or the software CCF causes spurious actuation. The evaluation uses realistic assumptions regarding initial operating conditions and assumes continued operability of the RCPs (except the events in which the event initiator is loss of power to the reactor coolant pumps (RCPs) or the actual failure of the RCPs) and the control systems. The qualitative evaluation results are compared to the acceptance criteria for each event initiator.

The results of the qualitative evaluation analyses are presented in the CCF Coping Analysis Technical Report.

b. Quantitative analysis

A detailed, quantitative analysis using qualified computer programs is conducted for the event requiring further detailed quantitative analyses in order to determine their compliance with the acceptance criteria.

The results of the quantitative evaluation analyses are presented in the CCF

SRM on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs" Item II.Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems," U.S. Nuclear Regulatory Commission, July 21, 1993.

7.8.4

No comI

7.8.5 References

1. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

2. ~~SECY-93-087, "Policy, Technical, and Licensing Issue Preparing to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," U.S. Nuclear Regulatory Commission , April 2, 1993.~~

3. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety-Related," U.S. Nuclear Regulatory Commission, 1985.

2) Bypassed or inoperable status indication – Regulatory Guide (RG) 1.47 (Reference 10)

3) Accident monitoring instrumentation – RG 1.97 (Reference 11)

4) Alarms for credited manual operator actions – ~~SECY-93-087~~ SRM on SECY-93-087 (Reference 12)

5) Coping with common-cause failures – ~~SECY-93-087 and (BTP) 7-19~~ SRM on SECY-93-087, and BTP 7-19 (Reference 13)

6) Manual initiation of protective actions – RG 1.62 (Reference 14)

7) Safe shutdown from outside the MCR – GDC 19 of 10 CFR Part 50, Appendix A (Reference 15)

8) Computerized procedures – Section 1 of DI&C-ISG-05 (Reference 16)

9) Technical support center – NUREG-0696 (Reference 17) and NUREG-0737, Supplement 1 (Reference 18)

10) Emergency operations facility – NUREG-0737, Supplement 1 (Reference 18)

These requirements are reflected in the APR1400 basic HSI, the APR1400 HSI, and the APR1400 HSI facilities, as applicable.

18.7.2.2 Concept of Operations

The concept of operations considers the following items and is developed and used during the HSI design process:

a. Crew composition

b. Roles and responsibilities of individual crew members

c. Personnel interaction with plant automation

The DMA switches provide the operator with the capability to actuate the engineered safety features (ESF) systems from the main control room (MCR). The DMA switches are diverse from the manual and automatic logic functions performed by digital equipment in the PPS and ESF-CCS.

c. GDC 19, “Control Room”

The MCR safety console is equipped with manual reactor trip initiation switches, manual ESF actuation switches and PPS operator modules (OMs) shared with the ESF-CCS and core protection calculator system (CPCS). Monitoring of the plant is accomplished through the use of the qualified indication and alarm system – P (QIAS-P), qualified indication and alarm system – non-safety (QIAS-N) and information processing system (IPS) displays. The DAS (including DPS, DMA switches, and DIS) equipment are provided to protect against a DBE concurrent with a postulated CCF in the safety I&C systems.

d. GDC 21, “Protection System Reliability and Testability”

The DAS is designed to meet the reliability goal of the plant I&C systems.

e. GDC 22, “Protection System Independence”

The independence between the DAS and the protection systems conforms to the independence requirements of IEEE Std 384-1992 (Reference 8) and IEEE Std 603-1991 (Reference 4).

f. GDC 24, “Separation of Protection and Control System”

The electrical, physical and communication isolations are maintained between the safety I&C systems and the DAS which is a non-safety system.

Where safety sensors are shared between the DAS and the safety I&C systems, the qualified isolators in the auxiliary process cabinet–safety (APC-S) prevent adverse interaction with the safety functions induced by DAS failures.

g. GDC 29, “Protection Against Anticipated Operational Occurrences”

Plant initiating events have been analyzed and the safety I&C systems protect the plant against AOO. The DAS, which is diverse from the safety I&C systems and not subject to CCF in the safety I&C systems, provides backup safety functions for AOO.

SRM on SECY-93-087,

3.3 Regulatory Guidances and Reports

3.3.1 ~~SECY-93-087,~~ “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” ~~1993,~~ Item II.Q, “Defense against Common-Mode Failures in Digital Instrumentation and Control Systems”, ~~and the associated Staff Requirements Memorandum (SRM), 1993~~ (Reference 7).

Design features for D3 for the PPS and ESF-CCS are implemented in accordance with SRM on SECY-93-087, as referenced by NUREG-0800.

July 1993

The DAS is designed to comply with the requirements of defense against a postulated CCF in the protection systems.

8 References

1. APR1400-E-J-NR-14001-P, "Component Interface Module", Rev. 0, November 2014

2. APR1400-Z-A-NR-14019, "CCF Coping Analysis", Rev. 0, November 2014

3. APR1400-Z-J-NR-14001-P, "Safety I&C System", Rev. 0, November 2014

4. IEEE Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

5. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants"

6. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment that is not Safety-Related", April 1985

7. ~~SECY 93-087,~~ "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", ~~April 1993, and the associated Staff Requirements Memorandum,~~ July 1993

8. IEEE Std 384-1992, "IEEE Standard Criteria for Indep

9. NUREG-0800, "Standard Review Plan," Chapter 7, B
and Defense-in-Depth in Digital Computer-Based Inst

10. NUREG-0800, "Standard Review Plan," Chapter 18, Appendix 18-A, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses"

11. ANSI/ANS 58.8-1994, "Time Response Design Criteria for Safety-Related Operator Actions"

12. NUREG-0711, "Human Factors Engineering Program Review Model", Rev. 2, February 2004

13. NUREG/CR-6303, "Method for Performing Diversity and Defense-in Depth Analyses of Reactor Protection Systems", October 1994

14. APR1400-Z-J-NR-14003-P, "Software Program Manual", Rev. 0, November 2014

15. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

16. APR1400 DC Quality Assurance Manual

17. APR1400-K-Q-TR-11005-N, "KHNP Quality Assurance Program Description for the APR1400 Design Certification"

18. ANSI/ANS-58.11-1995 (R2002), "Design Criteria for Safe Shutdown following Selected Design Basis Events in Light Water Reactors"

SRM on SECY-93-087,

Item II.Q, "Defense against Common-Mode Failures in Digital Instrumentation and Control Systems",

The ESF-CCS performs the ESF component cooling water and essential service water functions and executes component control through the interfacing ESFAS portion of the PPS. The ESF-CCS performs selective 2-out-of-4 coincidence logic for the four-division ESFAS initiation signals derived from the PPS and component control logic of ESF components.

SRM on SECY-93-087,

3.3 Staff Requirements Memorandum and NUREG Reports

3.3.1 ~~SECY 93-087,~~ "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors", ~~II.Q,~~ "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems"

Item II.Q,

Analyses and design features for D3 for the safety I&C system are provided in accordance with SECY 93-087, II.Q, as referenced by NUREG-0800.

The DPS automatically initiates a reactor trip on high containment pressure to assist the mitigation of the effects of a postulated CCF of the safety I&C system, concurrent with a main steam line break inside containment. The DPS also automatically initiates a safety injection actuation signal (SIAS) on low pressurizer (PZR) pressure in case of loss of coolant accident with CCF of the safety I&C system.

The DMA switches are provided to allow manual control capability to support ESF actuation in the event of a postulated CCF of the safety I&C system.

The DIS displays position 4 variables defined in Staff Requirements Memorandum (SRM) on SECY 93-087 in the event of a postulated CCF of the safety I&C system.

Compliance with SRM to SECY 93-087 and the diverse I&C system design features are addressed in the Diversity and Defense-in-Depth TeR.

The detailed CCF analysis methodology and the results are described in the CCF Coping Analysis TeR.

3.3.2 SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors", II.T, "Control Room Annunciator (Alarm) Reliability"

The alarm systems are designed to meet the requirements of the SRM to SECY 93-087, Item II.T and are implemented in both the IPS and QIAS-N, and are designed as independent and diverse from each other. Therefore, the implemented alarm functions have redundancy and diversity features in the alarm system as specified in SECY-93-087, Item, II.T.

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety grade. The QIAS-N processors also provide redundant processing in a hot standby configuration. Multi-division information displayed by the QIAS-N is independently processed and displayed by the IPS. The QIAS-N receives the processed information from each division of four interface and test processors (ITPs) and alarms any discrepancies from its own corresponding multi-division information calculators. Therefore, the implemented alarm function complies with SECY 93-087, Item II.T redundancy requirement.

The IPS and QIAS-N in which the alarm function is implemented are designed as independent and diverse from each other. The IPS receives alarm signals through fiber optic data link using unidirectional Ethernet via MTP from the PPS and ESF-CCS. The QIAS-N also receives alarm signals through unidirectional SDL via each division of ITP from the PPS and ESF-CCS. Therefore, the alarm functions by the IPS and QIAS-N would not impact the safety systems such as the PPS or ESF-CCS as well as the performance of the QIAS-N functions.

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:	**7.01**
Date of RAI Issue:	**06/22/2015**

Question No. 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.

Response

The alarm system is designed to meet the Staff Requirements Memorandum (SRM) on SECY-93-087, Item II.T. To clarify, related DCD and Technical Report sections will be revised as indicated in the attachment associated with this response.

The alarm system conforms to the SRM on SECY-93-087, Item II.T as follows:

Redundancy

The alarm system is implemented a one of the information processing system (IPS) servers. This alarm system is arranged in an active/standby redundant configuration, and provides processing functionality to support the needs of alarm processing. The alarm system transmits processing results to multiple operator consoles as well as the large display panel (LDP) in the

MCR main control room (MCR) through redundant data communication network - information (DCN-I) networks.

In addition, the qualified indication and alarm system-non-safety (QIAS-N) system provides alarm processing functionalities by diverse hardware and software from the IPS. The QIAS-N system has redundant configuration for the servers and networks.

The MCR provides multiple operator consoles, and each operator console has four (4) information flat panel displays (IFPDs) to display not only plant operation displays but also alarm displays. The LDP also provides not only alarms for essential components and process variables but also system group alarms.

The safety console provides alarm indications on the QIAS-N flat panel display (FPD) and mini-LDP through redundant QIAS-N servers and network.

Independence and Separation

The alarm system which is one server of IPS servers has same design configuration of independence and separation as the IPS. The communication independence and electrical isolation for the IPS is designed such that this design can meet the requirements per IEEE Std. 603, 384 and IEEE Std. 7-4.3.2.

For alarm data of safety systems, the IPS only receives alarm data from the safety systems through unidirectional communication via maintenance and test panel (MTP) and distributed control system (DCS) gateway. Fiber-optic isolation is used to ensure electrical isolation between the MTP in safety side and the DCS gateway in non-safety side. No the IPS sends any data to the safety systems.

The QIAS-N servers also receive alarm data from the safety systems through unidirectional fiber optic serial data link (SDL) via interface and test processor (ITP). Fiber-optic isolation is used to ensure electrical isolation between the ITP and QIAS-N severs.

So, any failures from the IPS do not adversely affect to perform the safety function in the plant protection system (PPS) and engineered safety features-component control system (ESF-CCS).

Safety-Related Alarm System Requirements

Since all APR1400 safety actuation functions are automatic, there are no safety-related alarms that should be designed to class 1E equipment and circuit.

Impact on DCD

DCD Tier 2, Subsections 7.1.2.37, 7.1.5, and 7.5.5, and Table 7.1-1 will be revised as indicated in the Attachment.

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.

Safety I&C System Technical Report APR1400-Z-J-NR-14001-P/NP, Subsection 3.3.2 will be revised as indicated in the Attachment.

SRM on SECY-93-087

7.1.2.33 Conformance with GDC 38

The I&C systems that are applicable to GDC 38, as shown in Table 7.1-1, are designed in accordance with GDC 38.

7.1.2.34 Conformance with GDC 41

The I&C systems that are applicable to GDC 41, as shown in Table 7.1-1, are designed in accordance with GDC 41.

7.1.2.35 Conformance with GDC 44

The I&C systems that are applicable to GDC 44, as shown in Table 7.1-1, are designed in accordance with GDC 44.

7.1.2.36 Conformance with SECY-93-087, Item II.Q

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). The analyses and design features address postulated safety system CCFs and are described in the Diversity and Defense-in-Depth Technical Report.

7.1.2.37 ~~Conformance with SECY-93-087, Item II.T~~

Conformance with SRM on SECY-93-087, Item II.T

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). The APR1400 design complies with the requirements as follows:

SRM on SECY-93-087

a. Redundancy

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety.

Major equipment of the IPS such as the computational server, alarm server, historical data storage and retrieval (HDSR) server, and data communication are configured to primary and standby processors.

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~~ SRM on 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~~ July 21, 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.

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.

SRM on SECY-93-087, Item II.T

Table 7.1-1 (3 of 6)

Applicable Criteria		Title	I&C System							Section in APR1400 DCD
			RTS	ESF System	QIAS-P	QIAS-N	PCS	P-CCS	DAS	
Staff Requirements Memoranda										
35	SECY-93-087, Item II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems	×	×					×	7.2, 7.3, 7.8, 7.9
36	~~SECY-93-087, Item II.T~~	Control Room Annunciator (Alarm) Reliability				×				7.5, 7.9
NRC Regulatory Guides										
37	NRC RG 1.22	Periodic Testing of Protection System Actuation Functions	×	×						7.2, 7.3,, 7.9
38	NRC RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	×	×		×				7.2, 7.3, 7.5, 7.6, 7.9
39	NRC RG 1.53	Application of the Single-Failure Criterion to Safety Systems	×	×	×					7.2, 7.3, 7.4, 7.5, 7.6, 7.9
40	NRC RG 1.62	Manual Initiation of Protective Actions	×	×					×	7.2, 7.3, 7.8
41	NRC RG 1.75	Criteria for Independence of Electrical Safety Systems	×	×	×	×	×	×	×	7.2, 7.3, 7.4, 7.5, 7.6, 7.7, 7.8, 7.9
42	NRC RG 1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants			×	×				7.5
43	NRC RG 1.105	Setpoints for Safety-Related Instrumentation	×	×	×	×				7.2, 7.3, 7.4, 7.5, 7.6, 7.9
44	NRC RG 1.118	Periodic Testing of Electric Power and Protection Systems	×	×	×	×				7.2, 7.3, 7.4, 7.5, 7.6, 7.9
45	NRC RG 1.151	Instrument Sensing Lines	×	×	×					7.2, 7.3, 7.5,
46	NRC RG 1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants	×	×	×					7.2, 7.3, 7.5, 7.9

7. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," Rev. 1, U.S. Nuclear Regulatory Commission, February 2010.

8. IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2002.

9. IEEE Std. 384-1992, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers, 1992.

10. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," Rev. 3, U.S. Nuclear Regulatory Commission, April 1995.

11. 10 CFR 50.34(f)(2)(xviii), "Instrumentation for Detection of Inadequate Core Cooling," [II.F.2], 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.

13. ~~SRM to SECY-93-087~~ SRM on SECY-93-087, Item II.T, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advance Light-Water Reactor (ALWR) Designs," U.S. Nuclear Regulatory Commission, ~~April 2~~ July 21, 1993.

14. 10 CFR 50.34(f)(2)(iv), "Safety Parameter Display Console" [I.D.2] U.S. Nuclear Regulatory Commission.

15. 10 CFR 50.34 (f)(2)(xxv), "Additional TMI-related Requirements," [III.A.1.2], U.S. Nuclear Regulatory Commission.

16. 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.

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

18. 10 CFR 50.34(f)(2)(xi), "Direct Indication of Relief and Safety Valve Position (Open or Closed) in the Control Room," [II.D.3], U.S. Nuclear Regulatory Commission.

The ESF-CCS performs the ESF component cooling water and essential service water functions and executes component control through the interfacing ESFAS portion of the PPS. The ESF-CCS performs selective 2-out-of-4 coincidence logic for the four-division ESFAS initiation signals derived from the PPS and component control logic of ESF components.

3.3 Staff Requirements Memorandum and NUREG Reports

3.3.1 SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors", II.Q, "Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems"

Analyses and design features for D3 for the safety I&C system are provided in accordance with SECY 93-087, II.Q, as referenced by NUREG-0800.

The DPS automatically initiates a reactor trip on high containment pressure to assist the mitigation of the effects of a postulated CCF of the safety I&C system, concurrent with a main steam line break inside containment. The DPS also automatically initiates a safety injection actuation signal (SIAS) on low pressurizer (PZR) pressure in case of loss of coolant accident with CCF of the safety I&C system.

The DMA switches are provided to allow manual control capability to support ESF actuation in the event of a postulated CCF of the safety I&C system.

The DIS displays position 4 variables defined in Staff Requirements Memorandum (SRM) on SECY 93-087 in the event of a postulated CCF of the safety I&C system.

Compliance with SRM to SECY 93-087 and the diverse I&C system design features are addressed in the Diversity and Defense-in-Depth TeR.

The detailed CCF analysis methodology and the results are described in the CCF Coping Analysis TeR.

SRM on SECY-93-087

3.3.2 ~~SECY 93-087~~, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors", ~~II.T~~, "Control Room Annunciator (Alarm) Reliability"

Item II.T

The alarm systems are designed to meet the requirements of the SRM to SECY 93-087, Item II.T and are implemented in both the IPS and QIAS-N, and are designed as independent and diverse from each other. Therefore, the implemented alarm functions have redundancy and diversity features in the alarm system as specified in ~~SECY 93-087, Item, II.T~~.

SRM on SECY-93-087, Item II.T

The alarm systems are implemented in the software driven IPS and QIAS-N. The alarm functions in the IPS and QIAS-N are non-safety grade. The QIAS-N processors also provide redundant processing in a hot standby configuration. Multi-division information displayed by the QIAS-N is independently processed and displayed by the IPS. The QIAS-N receives the processed information from each division of four interface and test processors (ITPs) and alarms any discrepancies from its own corresponding multi-division information calculators. Therefore, the implemented alarm function complies with ~~SECY 93-087, Item II.T~~ redundancy requirement.

SRM on SECY-93-087, Item II.T

The IPS and QIAS-N in which the alarm function is implemented are designed as independent and diverse from each other. The IPS receives alarm signals through fiber optic data link using unidirectional Ethernet via MTP from the PPS and ESF-CCS. The QIAS-N also receives alarm signals through unidirectional SDL via each division of ITP from the PPS and ESF-CCS. Therefore, the alarm functions by the IPS and QIAS-N would not impact the safety systems such as the PPS or ESF-CCS as well as the performance of the QIAS-N functions.

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:	**07.01**
Date of RAI Issue:	**06/22/2015**

Question No. 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 requirements 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 prescribed 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.

Response

APR1400 FSAR, Tier 2, Section 7.2 describes the “Reactor Trip System” which includes the ENFMS and the APC-S. Subsection 7.2.3.6 “Analysis” states conformance with IEEE Std.603 and refers to the “Safety I&C System” Technical Report (APR1400-Z-J-NR-14001-P) for description of compliance. The ENFMS and APC-S are described as Safety I&C Systems in Subsection 4.1.1.5 and 4.1.1.6 of the Technical Report, and conformance with IEEE Std. 603-1991, including Clauses 5.1, 5.3, 5.5, and 5.6 for all Safety I&C Systems including ENFMS and APC-S, is addressed in Appendix A, “Conformance to IEEE Std. 603-1991” of the Technical Report .

APR1400 FSAR, Tier 2, Subsection 7.3.1 states “The engineered safety features (ESF) system consists of four channels of sensors, and the auxiliary process cabinet-safety (APC-S), and for divisions of the engineered safety features actuation system (ESFAS) portion of the plant protection system (PPS), the safety portion of radiation monitoring system (RMS), and the engineered safety features-component control system (ESF-CCS)”.

The RMS consists of two (2) channels; the Safety Related Divisionalized Cabinet (SRDC) and the non-safety related RMS computer cabinet, as shown in Figure 7.3-23. The safety portion of the RMS consists of the radiation element, the local unit, and the SRDC. The divisional SRDC transmits the ESFAS initiation signals to the dedicated ESFAS measurement channels, as described in Subsection 7.3.1.1. The safety portion of RMS is part of the ESF system as described in the Subsection 7.1.1.3

Therefore, the safety portion of the RMS is a part of ESF system and designed to comply with the ESF System applicable criteria in the APR1400 FSAR, Tier 2, Table 7.1-1, “Regulatory Requirements Applicability Matrix”. The ESF system, including the safety portion of the RMS, complies with the requirements of IEEE Std.603-1991, Clauses 5.1, 5.3, 5.5, and 5.6, as described in Subsection 7.3.3.2 “Conformance with IEEE Std. 603”. Conformance to IEEE Std. 603 is addressed in the Appendix A, “Conformance to IEEE Std.603-1991” of the “Safety I&C System” Technical Report.

There are no other standalone safety-related I&C systems other than ENFMS, APC-S, and the SRDC in the RMS.

Impact on DCD

There is no impact on the DCD.

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 Reports.

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:	**Section 7.1**
Date of RAI Issue:	**06/22/2015**

Question No. 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?

Response

APR1400 DCD, Tier 2, Subsection 10.2.2.1, “General Description“ states, “The TGCS uses a digital monitoring and control system that controls the turbine speed, load, and flow for startup and normal operations. The control system operates the turbine MSVs, CVs, ISVs, and IVs. T/G supervisory instrumentation is provided for operational analysis and malfunction diagnosis.” Subsection 10.2.2.3 “Control and Protection,” provides more detailed information regarding the turbine/generator (T/G) I&C system.

APR1400 DCD, Tier 2, Section 10.2.5, “Combined License Information,” states in COL 10.2(1), “The COL applicant is to identify the turbine vendor and model” and in COL 10.2(2), “The COL applicant is to identify how the functional requirements for the overspeed protection system are met and provide a schematic of the TGCS and protection systems from sensors through valve actuators.” The COL applicant identifies the turbine vendor and model. Therefore, additional detailed information and structure depend on the vendor’s design features.

The turbine control system (TCS) interfaces with the plant protection system (PPS) in the safety I&C systems for the turbine trip function on a reactor trip. APR1400 DCD, Tier 2, Subsection 7.2.1.4, Item l and Figure 7.2-14 provide information about the turbine trip function and functional logic. The PPS transmits the turbine trip signal via hardwired connection to the TCS when the reactor trip initiation signal is generated as indicated on the right side of Figure 7.2-14.

APR1400 non-safety stand-alone I&C systems include the TCS, seismic monitoring system (SMS), vibration monitoring system (VMS), NSSS integrity monitoring system (NIMS), and fixed in-core detector amplification system (FIDAS).

Non-safety standalone I&C systems of APR1400 have no interface with safety I&C systems with the exception of the TCS and NIMS, which receive signals via unidirectional hardwired connections from the PPS and ex-core neutron flux monitoring system (ENFMS), respectively.

As GDC 24 requires that the interconnection of the protection and control systems be limited to ensure that safety is not significantly impaired upon failure of any single control system component or channel, the PPS and ENFMS are designed in such a way that the PPS and ENFMS do not receive any signals from non-safety systems but only send signals to non-safety systems. Also, electrical isolation is provided in the PPS and ENFMS through isolation devices.

The physical separation is described in Section A.5.6 of the Safety I&C System Technical Report.

Table 07.01-19-1 summarizes interfaces between the non-safety standalone I&C systems with safety I&C systems.

Table 07.01-19-1 Interface Summary

Non-Safety Standalone I&C System	Safety I&C System	Signal Direction	Type of Interface	Independence Requirements of GDC 24 and IEEE Std.603,5.6.3
TCS	PPS	Safety→Non-safety (Unidirectional)	Hardwired connection	Independence requirements are applicable
SMS	N/A	N/A	No interface	N/A (No interface with safety I&C system)
VMS	N/A	N/A	No interface	N/A (No interface with safety I&C system)
NIMS	ENFMS	Safety→Non-safety (Unidirectional)	Hardwired connection	Independence requirements are applicable
FIDAS	N/A	N/A	No interface	N/A (No interface with safety I&C system)

Impact on DCD

There is no impact on the DCD.

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 Reports.