REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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

Docket No. 52-046

RAI No.:	**198-8208**
SRP Section:	**14.02 - Initial Test Program - Design Certification and New License Applicants**
Application Section:	**14.2.12.1.25**
Date of RAI Issue:	**09/04/2015**

Question No. 14.02-22

Demonstrate how the Ex-Core Neutron Flux Monitoring System (ENFMS) Test described in APR1400 FSAR Tier 2, Section 14.2.12.1.25 meets the requirements of Criterion XI of Appendix B to 10 CFR Part 50.

Criterion XI, “Test Control,” of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 states, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design requirements. RG 1.68, “Initial Test Programs for Nuclear Power Plants” provides guidance on the initial test program.

APR1400 FSAR Tier 2, Section 14.2.12.1.25 provides the initial test for the ENFMS. The staff reviewed this test and finds that additional information is required to determine whether this test meets the requirements of Criterion XI of Appendix B to 10 CFR Part 50. Specifically, the staff requests the applicant to address the following items described below.

1) Item 1.1 under “Objective” states that the objective is to “verify the proper functional performance of the [ENFMS].” Test Method Item 3.1 states, “Using appropriate test instrumentation, simulate and vary input signals to the startup, safety, and control channels of the ex-core neutron flux monitoring system.” In addition Acceptance Criteria Item 5.1 states, “The [ENFMS] performs as described in Subsection 7.7.1.1 h.” It is unclear to the staff whether this test will verify the functional performance of all channels of the ENFMS or only the non-safety related channels since Subsection 7.7.1.1 h only describes the non-safety related portion of the ENFMS. Clarify this discrepancy.

2) Item 2.0, “Prerequisites” does not specify whether the factory acceptance testing need to be complete prior to this test. Clarify this issue.

3) RG 1.68, Section A-1.j, “Instrumentation and Control Systems” specify that “tests should be conducted, as appropriate, to verify redundancy and electrical independence.” Since the ENFMS contains both safety and non-safety related channels, what tests are performed to verify sufficient electrical independence between the safety and non-safety related channels of the ENFMS?

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question is within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

1) This test will verify the functional performance of all channels of the ENFMS, so the sentence in 5.1 of the Acceptance Criteria will be revised to add Subsection 7.2.1.1.c to 7.7.1.1.h.

2) Performance of factory acceptance testing (FAT) by the vendor before the equipment is delivered to the site is generally a requirement of the procurement specification and would be applicable to the ex-core neutron flux monitoring system as well. In addition, RG 1.68 does not require FAT before preoperational testing. Therefore, FAT is not an essential prerequisite to add into the procedure.

3) For redundancy, ENFMS consists of four redundant safety channels, two redundant startup channels, and two redundant control channels. For electrical independence, safety and non-safety channels are designed to meet RG 1.75 and IEEE Std. 384 by providing sufficient electrical isolation and physical separation. In detail, the circuits of safety and non-safety channels are physically separated in the safety channel drawer and the startup/control channel drawer respectively for physical separation, and the qualified isolation devices are applied to connections of Class 1E and non-Class 1E circuits for electrical isolation.

   For electrical independence between the safety and non-safety related channels of the ENFMS, testing, analysis or combination of testing and analysis will be performed as described in 3.b of Table 2.5.1-5 in Section 2.5.1.2 of the APR1400 FSAR Tier 1.

Impact on DCD

Section 14.2.12.1.25 of the APR1400 FSAR Tier 2 will be revised as indicated in the attached markup.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specification.

Impact on Technical/Topical/Environmental Reports

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

2.2 Ex-core neutron flux monitoring system instrumentation has been calibrated.

2.3 External test equipment has been calibrated and is operational.

2.4 Support systems required for operation of the ex-core neutron flux monitoring system are operational.

3.0 TEST METHOD

3.1 Using appropriate test instrumentation, simulate and vary input signals to the startup, safety, and control channels of the ex-core neutron flux monitoring system.

3.2 Monitor and record all output signals as a function of variable inputs provided by test instrumentation.

3.3 Record the performance of audio and visual indicators in response to changing input signals.

4.0 DATA REQUIRED

4.1 Values of input and output signals for correlation purposes, as required

4.2 Values of all output signals triggering audio and visual alarms

5.0 ACCEPTANCE CRITERIA

5.1 The ex-core neutron flux monitoring system performs as described in ~~Subsection 7.7.1.1 h~~.

Subsections 7.2.1.1.c and 7.7.1.1.h.

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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

Docket No. 52-046

RAI No.:	**198-8208**
SRP Section:	**14 – Verification Test Program**
Application Section:	**14.2**
Date of RAI Issue:	**09/04/2015**

Question No. 14.02-24

Demonstrate how the Reactor Regulating System (RRS) Test described in APR1400 FSAR Tier 2, Section 14.2.12.1.28 meets the requirements of General Design Criterion (GDC) 1 of Appendix A to 10 CFR Part 50.

GDC 1, “Quality standards and records” of Appendix A, “General Design Criteria for Nuclear Power Plants” to 10 CFR Part 50 states, in part, that structures, systems, and components important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed.

APR1400 FSAR Tier 2, Section 14.2.12.1.28 describes the initial test for the RRS. The staff reviewed this test and finds that additional information is required regarding the pre-requisites for the RRS test. Specifically, Prerequisite Item 2.2 states, “RRS software is installed and instrumentation has been calibrated.” The staff reviewed the tests proposed in the initial test program for other software-based instrumentation and control (I&C) systems and did not find software installation a prerequisite for these systems. Clarify whether software installation should be a prerequisite for those other I&C system tests.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question are within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

A prerequisite for completing construction activities includes software installation and therefore, the necessity to test the functions of the software driven systems. For clarification, the prerequisite for testing other major software-based instrumentation and control (I&C) systems will be revised to include software installation. Since the Ex-Core Neutron Flux Monitoring System (ENFMS) has no software, DCD Sections 14.2.12.1.25 will be revised. For the software installation of the Plant Protection System (PPS), the Engineered Safety Features-Component Control System (ESF-CCS), the Fixed In-core Detector Amplification System (FIDAS) and the Reactor Power Cutback System (RPCS), DCD Sections 14.2.12.1.23, 14.2.12.1.24, 14.2.12.1.26 and 14.2.12.1.32 will be revised.

Impact on DCD

DCD Subsection 14.2.12.1.32 will be revised as indicated in the Attachment. Changes to other sections were made as a result of the upgrade effort (ref. KHNP letter MKD/ NW-16-0156L, dated February 24, 2016; ML16056A003) and are included for information.

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 and Environmental Report.

1.5 To verify the operation of the maintenance and test panel/interface and test processor for ESF-CCS

1.6 To verify the proper indication and alarms for ESF-CCS

1.7 To verify the operation of the operator module for ESF-CCS

1.8 To verify RSR Transfer function

1.9 To verify the operation of the ESF-CCS power supplies

1.10 To demonstrate redundancy, electrical independence, coincidence, and safe failure on loss of power.

2.0 PREREQUISITES

2.1 Construction activities on the ~~engineered safety feature actuation system (~~ESFAS~~)~~ have been completed and system software is installed.

2.2 ESFAS instrumentation has been calibrated.

2.3 External test instrumentation is available and calibrated.

2.4 Support systems required for operation of the ESFAS are operational.

3.0 TEST METHOD

3.1 Energize power supplies and observe output voltages.

3.2 Simulate ground faults and observe operation of the ground fault detectors.

3.3 Individually de-energize each group relay and monitor contact operation.

3.4 Test manual trips and monitor relay operation.

2.2 PPS and ESF-CCS ~~system~~ software is installed.

2.3 PPS instrumentation has been calibrated.

2.~~3~~4 External test instrumentation is available and calibrated.

2.~~4 Support~~5 The interface systems ~~required~~ for ~~operation of the trip circuit breakers, ESF-CCS, and PPS~~PPS such as CPCS, RTSS and ESF-CCS are operational.

3.0 TEST METHOD

3.1 Energize power supplies and verify output voltage.

~~3.2 Simulate ground faults and observe operation of the ground fault detectors.~~

~~3.3~~3.2 Using simulated reactor trip signals, trip each reactor trip circuit breaker located in the RTSS with the breaker in the test position. Observe the reactor trip circuit breaker operation.

3.~~4~~3 Repeat Step 3.~~3~~2 with the reactor trip circuit breakers in the operate position.

3.~~5 Exercise~~4 Perform the bistable ~~comparators~~ logic test using ~~internal and external test circuitry~~the MTP and observe the setpoints used in the bistable logic and operation of the appropriate ~~ESFAS~~bistable logic.~~–~~

3.5 Perform the coincidence logic test using the MTP and observe the operation of the coincidence logic.

3.6 Check the operation of ~~trip channel~~operating bypass features ~~including, where applicable, observation of the~~ observe the setpoints at which the ~~trip~~operating bypasses are ~~canceled~~ automatically removed.

5.8 Bistable trip function operations should be as specified in the related design specification.

5.9 Interlock functions should operate as specified in the related design specification.

5.10 Operating bypass operation should be as specified in the related design specification.

5.11 Trip channel bypass operation should be as specified in the related design specification.

14.2.12.1.25 Ex-Core Neutron Flux Monitoring System

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To verify the proper functional performance of the ex-core neutron flux monitoring system

1.2 To verify the proper performance of audio and visual indicators

2.0 PREREQUISITES

2.1 Construction activities on the ex-core neutron flux monitoring system have been completed ~~and system software is intalled~~.

2.2 Ex-core neutron flux monitoring system instrumentation has been calibrated.

2.3 External test equipment has been calibrated and is operational.

2.4 Support systems required for operation of the ex-core neutron flux monitoring system are operational.

1.2 To verify the measured output is provided to the IPS through the data link server and DCS network.

1.3 Verify Cable Continuity.

1.4 Verify Cable Insulation Resistance.

2.0 PREREQUISITES

2.1 Construction activities on the in-core nuclear instrumentation system are complete ~~.~~ and system software is installed ~~(~~Detectors do not need to be installed ~~)~~.

2.2 Fixed in-core nuclear signal channel instrumentation has been calibrated.

2.3 External test equipment has been checked and calibrated.

2.4 Support systems required for operation of the in-core nuclear instrumentation system are operational.

3.0 TEST METHOD

3.1 Measure and record cabling insulation resistance.

3.2 Using external test instrumentation, simulate in-core detector signals into the signal conditioning circuits.

3.3 Using internal test circuits, test each amplifier for proper operation in accordance with manufacturer's instruction manual.

3.4 Vary the simulated inputs to the amplifier and record its values displayed by the information processing system.

4.0 DATA REQUIRED

4.1 Record values of all simulated inputs, appropriate intermediate values, and outputs. (The online test program automatically performs this task.)

5.0 ACCEPTANCE CRITERIA

5.1 The COLSS performs as described in Subsection 7.7.1.4.

5.2 The test result of COLSS test program should meet the acceptance criteria for each test case which is specified in related design documents.

14.2.12.1.32 Reactor Power Cutback System Test

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate proper operation of the ~~reactor power cutback system (~~RPCS~~)~~

2.0 PREREQUISITES

2.1 Construction activities on the RPCS have been completed.

2.2 RPCS instrumentation has been calibrated.

2.3 External test equipment has been checked and calibrated.

and RPCS software is installed.

2.4 Support systems required for the operation of the RPCS are operational.

3.0 TEST METHOD

3.1 Using actual or simulated interface inputs to the RPCS, observe receipt of these signals at the RPCS.

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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Docket No. 52-046

RAI No.:	**198-8208**
SRP Section:	**14.02 - Initial Plant Test Program - Design Certification and New License Applicants**
Application Section:	**14.02**
Date of RAI Issue:	**09/04/2015**

Question No. 14.02-30

Justify why operation of the ex-core neutron flux monitoring system(ENFMS) is not a prerequisite for the Internal Vibration Monitoring System (IVMS) Test.

General Design Criterion (GDC) 1, “Quality Standards and records” of Appendix A, “General Design Criteria for Nuclear Power Plants” to 10 CFR Part 50 states, in part, that structures, systems, and components important to safety shall be tested to qualify standards commensurate with the importance of the safety functions to be performed.

APR 1400 FSAR Tier 2, Section 14.2.12.1.41 provides the Initial test descriptions for the IVMS test. APR 1400 FSAR Tier 2, Section 7.7.1.5 “Nuclear Steam Supply System Integrity Monitoring System”, states that the IVMS monitors the motion of the reactor internals by using the ex-core neutron flux signals from the ENFMS detectors. However, the operation of the ENFMS is not required as a prerequisite for this test. Justify why the operation of the ENMFS is not required for this test.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question is within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

Neutron flux signals from ENFMS are not needed to check the IVMS function for pre-operational testing; rather, simulated neutron flux signals are used in accordance with Section 14.2.1.1 which states that simulated signals or inputs are used to demonstrate the full range of the systems that are used during normal operation. Since ex-core signals are only valid after fuel-loading and the test of IVMS, section 14.2.12.1.41, is conducted during pre-operational test before fuel loading, simulated neutron flux signals are made by signal generators as input to IVMS for the test. Therefore, the pre-operational test of IVMS does not require ENFMS operation as a prerequisite.

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 Report.

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Docket No. 52-046

RAI No.:	**198-8208**
Review Section:	**14.02 – Initial Plant Test Program – Design Certification and New License Applicants**
Application Section:	**14.2**
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Question No. 14.02-31

Justify why all the automatic functions performed by the diverse protection system (DPS) are not verified in the DPS test described in APR1400 FSAR Tier 2, Section 14.2.12.1.49.

General Design Criterion (GDC) 1, “Quality standards and records” of Appendix A, “General Design Criteria for Nuclear Power Plants” to 10 CFR Part 50 states, in part, that structures, systems, and components important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed.

APR1400 FSAR Tier 2, Section 14.2.12.1.49 provides the initial test description for the DPS. The objective of this test is to verify the proper operation of the DPS. However, the test methods for this test only verify the operation of the reactor trip switch system (RTSS) trip circuit breaker and operation of the alternate auxiliary feedwater actuation signals using simulated input signals. It is not clear to the staff whether the simulated signals will be injected into the DPS. The staff requests the applicant to clarify this in the test methods description of this section. In addition, APR1400 FSAR Tier 2, Section 7.8 and the referenced technical reports identify additional automatic safety actuation signals performed by the DPS. The staff requests the applicant to justify why these functions are not verified in this initial test.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods, and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question are within the scope of the upgrade effort. Therefore, KHNP will

address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

As described in Section 7.8.1.1 of DCD Tier 2, the major function of the diverse protection system (DPS) is the generation of reactor trip signal, turbine trip signal, auxiliary feedwater actuation signal (AFAS), and safety injection actuation signal (SIAS) when required. Since Section 14.2.12.1.49 of DCD Tier 2 (Revision 0) did not include the initial test description for all functions of the DPS, the test plan was upgraded as part of the program described in the original response. The plan for 14.2.12.1.49 was changed to include the automatic actuation signals of the DPS and was previously submitted to the NRC (refer to KHNP submittal MKD/NW-16-0156L dated February 24, 2016; ML16056A003). Subsequent review has determined that additional clarifications pertaining to the DPS signals, (in particular the turbine trip signal), and other editorial changes should be made to enhance the initial test program (ITP). Therefore, the previously submitted ITP is being changed to include the more detailed information.

Impact on DCD

DCD Tier 2, Section 14.2.12.1.49 will be revised as indicated in the Attachment to this response.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specifications.

Impact on Technical/Topical/Environmental Reports

There is no impact on the Technical/Topical/Environmental Report.

3.2 During preoperational post-core hot functional tests, perform a full transfer of control from the MCR and perform a controlled cooldown from the remote shutdown console.

4.0 DATA REQUIRED

4.1 RCS temperatures and pressures

5.0 ACCEPTANCE CRITERIA

5.1 The ~~ability to cool~~plant can be cooled down ~~using remote~~and stabilized in the cold shutdown ~~instrumentation and controls has been demonstrated~~condition within the design limits from outside the main control room.

5.2 The remote shutdown console performs as described in ~~Subsection 7.4.1~~the related design specification.

14.2.12.1.49 Diverse Protection System Test

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate the proper operation of the diverse protection system (DPS)

the

1.2 To verify operation of DPS alarms and indications

1.3 To verify for each DPS channel, the channel bypass logic will ~~not produce initiation signals~~ while the DPS is in "Bypass".

block its trip outputs

1.4 To verify the operation of coincidence logic for steam generator 1 low level, steam generator 2 low level, high pressurizer pressure, low pressurizer pressure, high containment pressure, reactor trip on turbine trip, and DPS manual reactor trip signals.

Delete period.

1.5 To verify that the DPS trip and pretrip setpoints have been installed properly, and are within the acceptable tolerances.

Delete period.

1.6 To verify the ~~auxiliary feedwater initiation and safety injection initiation~~ initiation of reactor trip, turbine trip, auxiliary feedwater actuation, and safety injection actuation signals

1.7 To verify the DPS response time

2.0 PREREQUISITES

2.1 Construction activities on the ~~reactor trip switch system (~~RTSS~~)~~ and the DPS have been completed.

2.2 DPS instrumentation has been calibrated.

2.3 External test instrumentation is available and calibrated.

2.4 Support systems required for operation of the RTSS and DPS are operational.

3.0 TEST METHODS

input signals into the DPS

3.1 Energize power supplies and verify output voltage.

the

3.2 Using simulated ~~signals~~, trip each reactor trip circuit breaker with the breaker in the test position. Observe RTSS trip circuit breaker operation.

into the DPS

the initiation of

3.3 Using simulated input signals, observe ~~alternate~~ auxiliary feedwater actuation ~~signals~~signal, and safety injection actuation signal.

3.4 Observe the turbine trip signal which is automatically generated with a 3-second time delay after the initiation of DPS reactor trip signal.

4.0 DATA REQUIRED

4.1 Power supply voltages

4.2 Resistance for ground fault detector operation

4.3 Trip setpoints

5.0 ACCEPTANCE CRITERIA

~~5.1 The DPS performs as described in Section 7.8.~~

5.1 Initiation ~~logic~~ and bypass ~~logic~~ for steam generator 1 auxiliary feedwater ~~valves are~~ performs as specified in the related design specification.

5.2 Initiation ~~logic~~ and bypass ~~logic~~ for steam generator 2 auxiliary feedwater ~~valves are~~ performs as specified in the related design specification.

5.3 Initiation ~~logic~~ and bypass ~~logic~~ for the high pressurizer pressure ~~trip are~~ performs as specified in the related design specification.

5.4 Initiation ~~logic~~ and bypass ~~logic~~ for the low pressurizer pressure trip ~~are~~ performs as specified in the related design specification.

5.5 Initiation ~~logic~~ and bypass ~~logic~~ for the high containment pressure ~~trip are~~ performs as specified in the related design specification.

5.6 Initiation ~~logic~~ and bypass ~~logic~~ for the reactor trip on turbine trip ~~are~~ performs as specified in the related design specification.

5.7 Initiation and bypass for the manual reactor trip ~~are~~ performs as specified in the related design specification.

5.8 Indication and alarms for DPS should operate as specified in the related design specification.

5.9 ~~DPS~~ response time should be within design ~~value~~ as specified in the related design specification.

5.10 Initiation and bypass logics for the turbine trip signal perform as specified in the related design specificati on.

logics

actuation signal caused by low steam generator level

reactor trip signal caused by

safety injection actuation signal caused by

reactor trip signal caused by

logics

the

System

limit

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Application Section:	**14.2**
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Question No. 14.02-32

Demonstrate how the operation of the low-level interlock is verified in the pre-core pressurizer performance test described in APR1400 FSAR Tier 2, Section 14.2.12.1.53.

General Design Criterion (GDC) 1, “Quality standards and records” of Appendix A, “General Design Criteria for Nuclear Power Plants” to 10 CFR Part 50 states, in part, that structures, systems, and components important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed.

APR1400 FSAR Tier 2, Section 14.2.12.1.53 provides the initial test description for the pre-core pressurizer performance test. The acceptance criterion for this test states that the pressurizer performs as described in Subsections 7.7.1 and 5.4.10. APR1400 FSAR Tier 2, Subsection 7.7.1.1 b. "Pressurizer pressure and level control systems," states that the control system has a low-level interlock and a high-pressure interlock. The low-level interlock shuts off all the heaters when the level falls below a setpoint. The staff could not identify where the proper response of the heaters to the pressurizer low-level interlock is verified in this test. As such, the staff requests the applicant to demonstrate how the response of the pressurizer heaters to this interlock is verified in this test.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question is within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the

upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

All interlocks related to the pressurizer pressure and level are verified by decreasing and increasing the pressurizer pressure and level through the pre-core pressurizer performance test as specified in 14.2.12.1.53.

Specifically, the pressurizer low-level interlock signal, which is the pressurizer low level heater cutoff signal, turns all pressurizer heaters off. Once pressurizer level drops below the pressurizer low level heater cutoff setpoint (as specified in test method 3.6), verification of the interlock is made by operator actions in attempts to turn all heaters on manually with the pressurizer low-level interlock signal activated. If all the heaters are not able to be turned on at these conditions, the pressurizer low-level interlock signal is considered valid.

Additionally, DCD Tier 2 14.2.12.1.53, test method 3.6 will be revised to clarify the specified level for the verification as indicated in Attachment 1.

Impact on DCD

DCD Tier 2 14.2.12.1.53, test method 3.6 will be revised as indicated in Attachment 1.

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.

3.3 Make a low level error in the pressurizer and observe letdown orifice isolation valve response alarm and interlock setpoints.

3.4 Make a high level error in the pressurizer and observe letdown orifice isolation valve response alarm and interlock setpoints.

3.5 Make a low pressurizer level and observe operation of the charging control valves.

3.6 ~~Make a low-low pressurizer level~~ and observe heater response and alarm and interlock setpoints.

Make pressurizer level below the low level interlock setpoint

4.0 DATA REQUIRED

4.1 Response of pressurizer heaters to actual pressure and level signals

4.2 Response of spray valves to actual pressurizer pressure

4.3 Response of charging control valves to actual pressurizer level

4.4 Response of letdown orifice isolation valves to actual low pressurizer level error

4.5 Values of parameters at which alarms and interlocks occur.

5.0 ACCEPTANCE CRITERIA

5.1 The pressurizer performs as described in Subsections 7.7.1 and 5.4.10.

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Application Section:	**14.2**
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Question No. 14.02-33

Demonstrate how the test objective to verify proper operation and sequencing of the control element drive mechanism (CEDM) is accomplished in the pre-core CEDM performance test described in APR1400 FSAR Tier 2, Section 14.2.12.1.54.

General Design Criterion (GDC) 1, “Quality standards and records” of Appendix A, “General Design Criteria for Nuclear Power Plants” to 10 CFR Part 50 states, in part, that structures, systems, and components important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed.

APR1400 FSAR Tier 2, Section 14.2.12.1.54 provides the initial test description for the pre-core CEDM performance test. Test objective 1.3 specifies the verification of the proper operation and sequencing of the CEDM. APR1400 FSAR Tier 2, Section 7.7.1.1 a., "Reactivity control systems," states that there are five modes of control: sequential group movement in manual and automatic control, manual group movement, manual individual CEA movement, and standby. The digital rod control system (DRCS) and Information Processing System (IPS) monitor proper sequential motion and provide an alarm for out-of-sequence conditions. The staff could not identify which test method will verify the proper sequencing of the CEDM. Provide a test method to verify this test objective.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question are within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by

February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

During the pre-core Control Element Drive Mechanism (CEDM) performance test in Tier 2, Section 14.2.12.1.54, the functions of the digital rod control system (DRCS) and the CEDM without CEA extension shaft are tested to verify that the CEDM operates in the proper power sequence with four CEDM coils for CEA insertion and withdrawal. In every insertion/ withdrawal step operation, the sequence of the CEDM coil power for each CEDM is monitored by the power regulator of the DRCS power cabinets. The test includes a check to ensure that the DRCS trouble alarm is actuated if abnormal CEDM power is sensed or CEDM motion is stopped. The recorded CEDM coil trace is also used to verify that the coil traces for withdrawal and insertion motion occur in the proper sequence. Verification of the proper sequencing will be added to the Acceptance Criteria in the ITP for 14.2.12.1.54.

For the proper operation of the regulating control groups, the out-of-sequence alarm is provided by the NSSS application program of the IPS. The out-of-sequence alarm, which is different than the above mentioned DRCS trouble alarm, is tested in Tier 2, Section 14.2.12.2.4 with the manual individual operation of each individual CEA when the other CEAs are positioned at the bottom of the core. For clarification, the out-of-sequence alarm test will be added to Tier 2, Section 14.2.12.2.4.

Impact on DCD

DCD Tier 2 Subsections 14.2.12.1.54 and 14.2.12.2.4 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

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

3.0 TEST METHOD

3.1 At the specified RCS temperature and pressure, measure and record the loop resistance for each of the CEDM coils.

3.2 Balance CEDM cooling system as required to maintain the coil temperatures within the specified limits.

3.3 Connect cabling between the reactor bulkhead and the DRCS cabinets and energize the CEDM. Measure and record the dc voltage across the upper gripper coil and across the shunt on the DRCS power switch assembly panel.

3.4 Operate the CEDM and observe count totalizer operation.

4.0 DATA REQUIRED

4.1 CEDM "cold" coil resistance

4.2 CEDM cable resistance

4.3 RCS temperature and pressure

4.4 CEDM coil loop resistance at specified RCS temperature and pressure

4.5 DC voltage across the upper gripper coil at the specified RCS temperature and pressure

4.6 DC voltage across the shunt , which indicates the CEDM coil power trace

4.7 CEDM count totalizer readings

5.0 ACCEPTANCE CRITERIA

5.1 The DRCS performs as described in Subsection 7.7.1.1 a.

5.2 The CEDM coil temperatures (calculated from the measured coil resistances) are less than the maximum allowable temperature of 177℃ (350℉).

5.3 The appropriate withdrawal and insertion coil traces occur in proper sequence.

14.2.12.1.55 Pre-Core Reactor Coolant System Flow Measurements

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To determine the pre-core reactor coolant system (RCS) flow rate

1.2 To establish baseline RCS pressure drops

2.0 PREREQUISITES

2.1 All permanently installed instrumentation has been properly calibrated and is operational.

2.2 All test instrumentation has been checked and calibrated.

2.3 RCS is operating at nominal hot zero-power (HZP) conditions.

2.4 Desired reactor coolant pumps (RCPs) are operating.

2.5 The core operating limit supervisory system (COLSS), core protection calculators (CPCs), and information processing system (IPS) are in operation.

3.0 TEST METHOD

3.1 RCS flow, pressure drops, and the data necessary to calculate RCS flows for four reactor coolant pump (RCP) operations are obtained.

5.5 IPS major/minor deviation alarm operates as designed.

5.6 CEA Deviation ~~alarm~~ operate as designed and out-of-sequence alarms

14.2.12.2.5 Post-Core Reactor Coolant and Secondary Water Chemistry Data

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To maintain the proper water chemistry for the RCS and ~~steam generators~~secondary system during post-core hot functional testing

1.2 To verify the adequacy of sampling and analysis procedures in establishing and maintaining proper chemistry

1.3 To establish baseline data for the RCS and the secondary system chemistry

2.0 PREREQUISITES

2.1 ~~Primary~~The primary and secondary sampling systems are operable.

2.2 Chemicals and test equipment to support hot functional testing are available.

2.3 The primary and secondary chemical addition systems are operable.

~~2.4 Purification ion exchangers are charged with resin.~~

3.0 TEST METHOD

~~3.1 Minimum sampling frequency for the steam generator and RCS is as specified by the chemistry manual.~~ 3.1 The sampling frequency is modified as required to provide reasonable assurance of the proper RCS and ~~steam generator~~secondary system water chemistry.

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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

Docket No. 52-046

RAI No.:	**198-8208**
SRP Section:	**14.2.12.2.9 – Post-Core Instrument Correlation**
Application Section:	**14.2.12.2.9**
Date of RAI Issue:	**09/04/2015**

Question No. 14.02-34

Demonstrate how the test objective of the Post-Core Instrument Correlation Test stated APR1400 FSAR Tier 2, Section 14.2.12.2.9 can be accomplished with the test methods described in this section.

Criterion XI, “Test Control,” of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 states, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design requirements. RG 1.68, “Initial Test Programs for Nuclear Power Plants” provides guidance on the initial test program.

APR1400 FSAR Tier 2, Section 14.2.12.2.9 provides the post-core instrument correlation test. The test objective states, “To demonstrate proper operation of the plant protection system (PPS), core protection calculators (CPCs), information processing system (IPS), and qualified indication and alarm system (QIAS).” However, the test methods only require the PPS, CP, IPS and QIAS readouts and the main control room instrument readings to be obtained. It does not appear that these test methods will demonstrate the proper operation of the PPS, CPC, IPS, and QIAS. In addition, the acceptance criteria for this test states, “The IPS, QIAS, PPS, and CPCs perform as described in Sections 7.2 and 7.7.” Sections 7.2 and 7.7 of the APR1400 FSAR Tier 2 contain a significant amount of design descriptions for these systems. It is unclear what specific design criteria need to be met for these systems with this test. As such, the staff requests the applicant to identify the specific acceptance criteria that need to be met with this test.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the

process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question is within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

The prerequisites for the Post-Core Instrument Correlation test are that PPS, CPCs, IPS, QIAS and COLSS are in operation, proper operation of those systems is demonstrated through the preoperational tests specified in the APR1400 FSAR Tier 2, Sections 14.2.12.1.24 for PPS, 14.2.12.1.31 for COLSS, 14.2.12.1.44 for IPS and QIAS, and 14.2.12.1.138 for CPCs.

For the Post-Core Instrument Correlation test, the operation of those systems is verified by simultaneously comparing the process variables which are displayed on those systems to the PCIC. The acceptance criteria of this test is the normal operation of those systems as described in Sections 7.2 and 7.7. Any abnormal operation which results in a display discrepancy between systems of the process variables listed in the revised ITP (ref. KHNP submittal MKD/NW-16-0156L dated February 24, 2016; ML16056A003), will be checked and corrected.

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 and Environmental Report.

REVISED RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

APR1400 Design Certification

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

Docket No. 52-046

RAI No.:	**198-8208**
SRP Section:	**14.02 - Initial Test Program - Design Certification and New License Applicants**
Application Section:	**14.02.12.2.11**
Date of RAI Issue:	**09/04/2016**

Question No. 14.02-35

Clarify whether the Post-Core Ex-Core Neutron Flux Monitoring System (ENFMS) Test described in APR1400 FSAR Tier 2, Section 14.2.12.2.11 is only applicable to the non-safety channels of the ENFMS.

Criterion XI, “Test Control,” of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 states, in part, that a test program shall be established to assure that all testing required to demonstrate that SSCs will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design requirements. RG 1.68, “Initial Test Programs for Nuclear Power Plants” provides guidance on the initial test program.

APR1400 FSAR Tier 2, Section 14.2.12.2.11 describes the post-core ENFMS test. The acceptance criteria for this test states that the “[ENFMS] performs as described in Subsection 7.7.1.1.h.” APR1400 FSAR Tier 2 Subsection 7.7.1.1 h., "Ex-core neutron flux monitoring system (non-safety channel)," only describes the operation of the non-safety channels of the ENFMS. As such, it is not clear to the staff whether this test is only applicable to the non-safety channels or whether it is also supposed to test the safety channels. The staff requests the applicant to clarify the scope of this test (i.e. whether it is for both safety and non-safety channels of the ENFMS or just the non-safety channels). In addition, if this test is only applicable to the non-safety channel of the ENFMS, the staff requests the applicant to identify where the safety channels of the ENFMS are tested in the post core initial test program.

Response

KHNP has reviewed the subject question and understands the staff’s request. KHNP is in the process of upgrading the test plans presented in Section 14.2 of the DCD. This effort is

focused on adding additional SSCs that are important to safety and risk significant as well as increasing the level of detail described in the DCD for test prerequisites, test methods and acceptance criteria for the various tests. It has been determined that the actions to be taken as a result of this question is within the scope of the upgrade effort. Therefore, KHNP will address the noted items in the upgrade effort, which is scheduled to be completed by February 1, 2016. A revised response to this question that incorporates the results of the upgrade effort will be submitted to the NRC after completion.

Response – (Rev. 1)

According to the RG 1.68, the proper functional performance tests of the ENFMS and the proper performance tests of the audio and visual indications for both safety and non-safety channels of ENFMS are conducted in the pre-operational testing described in 14.2.12.1.25.

During the initial fuel loading, neutron count rate is continuously monitored by displaying, recording, and audible information by two temporary source-range channels or at least one temporary channel and one permanent channel (startup channel of the ENFMS) as described in 14.2.10.1 so that all changes in the multiplication factor are observed. Before this step, the startup channels of the ENFMS are calibrated. For consistency, Section 14.2.10.1 will be revised to add this calibration.

The proper functional performance tests of the ENFMS and the proper performance tests of the audio and visual indications for both safety and non-safety channels of ENFMS are not required during post-core HFT in RG 1.68. Therefore, Section 14.2.12.2.11 in the APR1400 FSAR Tier 2 will be deleted.

As the Section 14.2.12.2.11 is deleted, the Table 14.2-7 in the APR1400 FSAR Tier 2 will be revised. For 2.g of RG 1.68, App. A, Subsection # and Individual Test will be revised to “14.2.10.1 Initial Fuel Loading”. For 4.c of RG 1.68, App. A, Subsection # and Individual Test will be revised to “14.2.10.2.1 Safe Criticality Criteria” because a minimum of 1 decade of overlap is observed between the startup and log safety channels of the ex-core nuclear instruments as described in g of the Section 14.2.10.2.1. For 4.d of RG 1.68, App. A, Subsection # and Individual Test will be revised to “14.2.12.1.24 Plant Protection System Test” because the operation of associated protective functions and alarms for plant protection is tested as described in the Section 14.2.12.1.24.

Impact on DCD

Section 14.2.10.1, Section 14.2.12.2.11, Table 14.2-2, and Table 14.2-7 of the APR1400 FSAR Tier 2 will be revised as indicated in the attached markup.

Impact on PRA

There is no impact on the PRA.

Impact on Technical Specifications

There is no impact on the Technical Specification.

Impact on Technical/Topical/Environmental Reports

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

The shutdown cooling system is in service to provide coolant circulation to provide reasonable assurance of adequate mixing and a means of controlling water temperature. The in-containment refueling water storage tank (IRWST) is in service and contains borated water at a volume and concentration conforming to the Technical Specifications. Applicable administrative controls are used to prevent unauthorized alteration of system lineups or change to the boron concentration in the reactor coolant system (RCS).

Minimum instrumentation for fuel loading consists of two temporary source-range channels installed in the reactor vessel or one temporary channel and one permanently installed ex-core nuclear channel in the event that one of the temporary channels becomes inoperative. Both temporary and permanent channels ~~are response checked with a neutron source.~~ The temporary channels display neutron count rate on a count rate meter installed in the containment and are monitored by personnel conducting the fuel loading operation. The permanent channel displays neutron count rate on a meter and strip chart recorder located in the main control room and is monitored by licensed operators. In addition, at least one temporary channel and one permanent channel are equipped with audible rate indicators in two locations, a temporary channel in the containment and a permanent or temporary channel in the main control room.

Continuous area radiation monitoring is provided during fuel handling and fuel loading operations. Permanently installed radiation monitors display radiation levels in the main control room and are monitored by licensed operators.

are required for calibration and for response check with a neutron source.

14.2.10.1.1 Safe Loading Criteria

Criteria for the safe loading of fuel require that loading operations stop immediately if:

a. The neutron count rate from either temporary nuclear channel unexpectedly doubles during any single loading step, excluding an anticipated change due to detector and/or source movement or spatial effects (i.e., fuel assembly coupling source with a detector).

b. The neutron count rate on any individual nuclear channel increases by a factor of 5 during any single loading step, excluding anticipated changes due to detector

4.2 ALMS alarm setpoints

4.3 RCS temperature and pressure

5.0 ACCEPTANCE CRITERIA

5.1 The ALMS ~~performs as described in Subsection 7.7.1.5~~shall performed leak/crack detection functions.

Section 14.2.12.2.11 will be deleted.

5.2 The ALMS alarm setpoints have been adjusted as necessary.

5.3 The baseline data of background noise shall be obtained

14.2.12.2.11 Post-Core Ex-Core Neutron Flux Monitoring System Test

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To verify the proper functional performance of the ex-core neutron flux monitoring system

1.2 To verify the proper performance of the audio and visual indicators

2.0 PREREQUISITES

2.1 Construction activities on the ex-core neutron flux monitoring system have been completed.

2.2 Ex-core neutron flux monitoring system instrumentation has been calibrated.

2.3 External test equipment has been calibrated and is operational.

2.4 Support systems required for the operation of the ex-core neutron flux monitoring system are operational.

2.5 Check source is available.

3.0 TEST METHOD

3.1 Using appropriate test instrumentation, simulate and vary input signals to the startup, safety, and control channels of the ex-core neutron flux monitoring system.

3.2 Monitor and record all output signals as a function of variable inputs provided by test instrumentation.

3.3 Record the performance of audio and visual indicators in response to changing input signals.

3.4 Using a check source, verify calibration of the startup, safety, and control channels.

4.0 DATA REQUIRED

4.1 Values of input and output signals for correlation purposes, as required

4.2 Values of all output signals triggering audio and visual alarms

4.3 Channel response to the check source

5.0 ACCEPTANCE CRITERIA

5.1 The ex-core neutron flux monitoring system performs as described in Subsection 7.7.1.1 h.

5.2 Safety channel operate as designed

5.3 Startup and control channel operate as designed

Section 14.2.12.2.11 will be deleted.

5.3 Interface between safety channel and startup and control channel should be satisfied

5.4 Boron Dilution Alarm System(BDAS) operate as designed

14.2.12.3 Low-Power Physics Test

Section 14.2.12.2.11 will be deleted.

14.2.12.3.1 Low-Power Biological Shield Survey Test

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To ~~measure~~ demonstrate the effectiveness of the radiation ~~in accessible locations of the plant outside the biological~~ shield

1.2 To obtain baseline levels for comparison with future measurements of radioactivity level buildup with operation

2.0 PREREQUISITES

2.1 Radiation survey instruments ~~are~~have been calibrated.

2.2 Background radiation levels have been measured in designated locations prior to initial criticality.

3.0 TEST METHOD

3.1 Measure gamma and neutron dose rates during low-power (<5 percent rated thermal power ~~[RTP]~~) operation~~.~~

4.0 DATA REQUIRED

4.1 Power level

4.2 Gamma and neutron dose rates at each specified location

Table 14.2-2

Post-Core Hot Functional Tests

Subsection	Test
14.2.12.2.1	Post-core hot functional test controlling document
14.2.12.2.2	~~Loose parts monitoring system~~NSSS Integrity Monitoring System (post-core)
14.2.12.2.3	Reactor coolant system flow measurements
14.2.12.2.4	Post-core control element drive mechanism performance
14.2.12.2.5	Post-core reactor coolant and secondary water chemistry data
14.2.12.2.6	Post-core pressurizer spray valve and control adjustments
14.2.12.2.7	Post-core reactor coolant system leak rate measurement
14.2.12.2.8	Post-core in-core instrumentation test
14.2.12.2.9	Post-core instrument correlation
14.2.12.2.10	Post-core acoustic leak monitor system test
~~14.2.12.2.11~~	~~Post-core ex-core neutron flux monitoring system test~~

Delete

Table 14.2-7 (13 of 18)

RG 1.68 APP. A	Subsection #	Individual Test
1.p.6	14.2.12.1.91 14.2.12.1.92	Containment polar crane test Fuel handling area cranes test
1.p.7	14.2.12.1.91 14.2.12.1.92	Containment polar crane test Fuel handling area cranes test
2.a	14.2.12.2.1	Post-core hot functional test controlling document
2.b	14.2.12.2.4	Post-core control element drive mechanism performance
2.c	14.2.12.1.24 14.2.12.2.4	Plant protection system test Post-core control element drive mechanism performance
2.d	14.2.12.2.7	Post-core reactor coolant system leak rate measurement
2.e	14.2.12.2.1 14.2.12.2.5	Post-core hot functional test controlling document Post-core reactor coolant and secondary water chemistry data
2.f	14.2.12.2.2 14.2.12.2.3 14.2.12.2.10	~~Loose parts~~NSSS integrity monitoring system (post-core) Reactor coolant system flow measurements Post-core acoustic leak monitor system test
2.g	~~14.2.12.2.11~~	~~Post-core ex-core neutron flux monitoring system test~~
2.h	14.2.12.2.8	Post-core in-core instrumentation test
3	14.2.10.1 14.2.12.3.5	Shutdown and regulating CEA group worth test Critical boron concentration test
4.a	14.2.12.3.2 14.2.12.3.5	Isothermal temperature coefficient test Critical boron concentration test
4.b	14.2.10.2.1	Shutdown and Safe Criticality Criteria
4.c	14.2.12.2.9 ~~14.2.12.2.11~~	Post-core instrument correlation ~~Post-core ex-core neutron flux monitoring system test~~
4.d	14.2.12.2.9 ~~14.2.12.2.11~~	Post-core instrument correlation ~~Post-core ex-core neutron flux monitoring system test~~
4.e	14.2.12.4.10 14.2.12.1.24 14.2.12.4.9	Steady-state core performance test Plant Protection System Test Biological shield survey test
4.f	14.2.12.1.106	Process and effluent radiological monitoring system test
4.g	14.2.12.4.4	Reactor coolant and secondary chemistry and radiochemistry test
4.h	14.2.12.4.9	Biological shield survey test