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.:	**524-8697**
SRP Section:	**14.2 – Initial Plant Test Program**
Application Section:	**14.02**
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Question No. 14.02-69

In APR1400 DCD Section 14.2, the applicant committed to RG 1.68, Revision 4. RG 1.68, Appendix A, “Initial Test Program,” states, in part, that it incorporates information relevant to applications for design certifications (DCs) under the applicable appendix to 10 CFR Part 52. For combined license (COL) applications under 10 CFR Part 52, the applicant must describe the ITP, in accordance with §52.79(a)(28). The requirements in 10 CFR 52.47 and 10 CFR 52.137 do not require a DC applicant to submit an ITP. However, DC applicants have previously submitted plans for an ITP in their applications to assist the COL applicant that references those DCs to meet the requirement in §52.79(a)(28) to include plans for an ITP in its COL application.

RG 1.68, Revision 4 provides guidance on initial fuel load and initial criticality tests in Appendix A, Section A-2, “Initial Fuel Load and Pre-Critical Tests,” and Section A-3, “Initial Criticality,” and initial fuel load and initial criticality test procedures in Appendix C, “Preparation of Procedures,” Section C-2, “Fuel Loading,” and Section C-3, “Initial Criticality Procedures.” However, the APR1400 DC application does not include any initial fuel load/initial criticality tests to conform to the guidance in RG 1.68, which specifies the following tests:

1. Initial Fuel Loading, to establish prerequisites and conditions for initial fuel loading and procedures to ensure safe loading

2. Inverse Count Ratio or 1/M Plot Test for Fuel Loading, for verification of sub-criticality during fuel loading

3. Initial Criticality, to describe the procedure for achieving initial criticality in a controlled manner

Since the above-described tests constitute a separate initial fuel load/initial criticality phase of the ITP, the DC applicant should consider creating a new table, similar to APR1400 DCD

Tables 14.2-1, “Preoperational Tests,” 14.2-2, “Post Core Hot Functional Tests,” 14.2-3, “Low-Power Physics Tests, and 14.2-4, “Power Ascension Tests” or consider renaming one of those tables, as appropriate.

To be consistent with the guidance in RG 1.68, Revision 4, the three initial fuel load/initial criticality tests noted above should be added to DCD Section 14.2 and listed in a new table of initial fuel load/initial criticality tests.

Response

An ITP for Initial Fuel Loading will be added before the Post-core Hot Functional Test and an ITP for Initial Criticality will be added before the Low Power Physics Test. The test plans for Initial Fuel Loading and Initial Criticality will be added to the current subsections rather than adding a new subsection(s) because adding a new subsection(s) would cause a complete restructuring of Section 14.2 in DCD Tier 2 and the four phase approach that the program is currently based upon and presented in the DCD. The title of the subsection 14.2.12.2 will be changed to denote that the section contains the Initial Fuel Loading as well as the Post Core Hot Functional Tests. Similarly, the title of the subsection 14.2.12.3 will be changed to denote that the section contains the Initial Criticality and Low-Power Physics Tests.

The Inverse Count Rate Ratio or 1/M Plot Test does not warrant a separate test since it is a continual process that is implemented during fuel loading and initial criticality. Therefore, the Inverse Count Rate Ratio or 1/M Plot Test will be described as a part of the Initial Fuel Loading process (refer to Test Method 3.9) and as part of the Initial Criticality process (refer to Test Method 3.2).

Tables 14.2-2 and 14.2-3 will be revised (refer to subsections indicated in Attachment 3) to incorporate the newly added ITPs and the titles of the tables will be revised in accordance with the new subsection titles. Table 14.2-7 (13 of 18) on compliance to RG 1.67 Appendix A will also be revised to incorporate the Initial Fuel Loading and Initial Criticality tests.

Impact on DCD

DCD Tier 2, Subsections 14.2.10.1.2 and 14.2.12.2.1 will be revised as indicated in Attachment 1.

DCD Tier 2, Subsection 14.2.12.3.1 will be revised as indicated in Attachment 2.

DCD Tier 2, Subsections 4.4.5.1, 14.2.7.1.13, 14.2.12.1.56, 14.2.12.2.2, 14.2.12.2.3, 14.2.12.2.4, 14.2.12.2.5, 14.2.12.2.6, 14.2.12.2.7, 14.2.12.2.8, 14.2.12.2.9, 14.2.12.2.10, 14.2.12.2.11, 14.2.12.3.2, 14.2.12.3.3, 14.2.12.3.4, 14.2.12.3.5, 14.2.12.3.6, Page ii and iv of the Table of Contents for Chapter 14, Table 14.2-2, Table14.2-3, and Table 14.2-7 (13, 14, 15 of 18) will be revised as indicated in 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

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

and/or source movement or spatial effects (i.e., fuel assembly coupling source with a detector).

A fuel assembly is not ungrappled from the refueling machine until stable count rates have been obtained. In the event that an unexplained increase in count rate is observed on any nuclear channel, the last fuel assembly loaded is withdrawn. The procedure and loading operation are reviewed and evaluated before proceeding to provide reasonable assurance of the safe loading of fuel.

14.2.10.1.2 Fuel Loading Procedure

An approved detailed test procedure is followed during the initial fuel loading to provide reasonable assurance that the evolution is completed in a safe and controlled manner. This procedure specifies applicable precautions and limitations, prerequisites, initial conditions, and the necessary procedural steps.

The test description for initial fuel loading is in subsection 14.2.12.2.1.

14.2.10.2 Initial Criticality

All systems required for startup or protection of the plant, including the plant protection system, safety injection system and containment spray system, are operable and in a state of readiness.

A predicted boron concentration for criticality is determined for the precritical CEA configuration specified in the procedure. This configuration requires all CEA groups to be fully withdrawn with the exception of the last regulating group, which remains far enough into the core to provide effective control when criticality is achieved. This position is specified in the procedure. The RCS boron concentration is then reduced to achieve criticality, at which time the regulating group is used to control the chain reaction.

Core response during CEA group withdrawal and RCS boric acid concentration reduction is monitored in the main control room by observing the change in neutron count rate as indicated by the permanent wide-range nuclear instrumentation.

Neutron count rate is plotted as a function of CEA group position and RCS boron concentration during the approach to criticality. Primary safety reliance is based on

3.6 Verify that displays respond correctly to actual or simulated input signals.

3.7 Verify the correct operation of the DIS switch on the Safety Console.

3.8 Verify the calculation of the heated-junction thermocouple (HJTC) heater power control signal.

3.9 Verify the control of the HJTC heater power via the DIS switch.

4.0 DATA REQUIRED

4.1 All simulated input signal values, appropriate intermediate values, and outputs

4.2 HJTC heater power control status receiving from the interfacing system (QIAS-P).

5.0 ACCEPTANCE CRITERIA

5.1 The DIS performs as described in Subsection 7.8.2.3.

5.2 The test results of the DIS should meet the acceptance criteria for each test case that is specified in related design document.

Initial Fuel Loading and Post-Core Hot Function Tests

14.2.12.2 ~~Post-Core Hot Functional Tests~~

Added Next pages as Initial Fuel Loading

~~14.2.12.2.1~~ Post-Core Hot Functional Test Controlling Document

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate the proper integrated operation of plant primary, secondary, and auxiliary systems with fuel loaded in the core

14.2.12.2.2

14.2.12.2.1 Initial Fuel Loading

1.0 OBJECTIVE

1.1 To provide a safe, organized method of accomplishing the initial fuel loading.

1.2 To establish the conditions under which the initial fuel loading is to be accomplished.

2.0 PREREQUISITES

2.1 All personnel shall read and understand the basic fuel load procedures.

2.2 Boron concentration shall be high enough to meet required shutdown margin.

2.3 The fuel loading evolution shall be controlled by use of approved plant procedure. The evolution is supervised by a licensed senior reactor operator.

2.4 Throughout “dry” core loading, the RCS water level shall be maintained above the top of the reactor vessel hot leg nozzle and below the vessel flange.

2.5 The proper seating and guide tube location of the two (2) neutron sources within their first cycle host assemblies shall be verified both before and after fuel loading.

2.6 The Integrated Head Assembly (IHA) and Upper Guide Structure (UGS) are removed from the reactor and stored.

2.7 The Reactor Coolant System (RCS) water quality has been verified to meet requirements of RCS chemistry and purity check.

2.8 The temporary fuel loading channel and auxiliary startup channels have been setup and calibrated.

2.9 At least one (1) permanent startup channel and one (1) temporary fuel loading channel are equipped with audible count rate indicators.

2.10 Neutron Response and Background Count Rates have been completed prior to the initiation of fuel loading. Channel Check prior to first fuel assembly loaded has been performed every 12 hours.

2.11 The collection of boron samples from the in-service SCS loop shall be started at one (1) hour intervals within 8 hours of commencement of initial fuel loading.

2.12 All fuel handling equipment pre-operation tests have been completed.

2.13 Continuous voice communications have been verified operational between the control room, refueling area in the containment building, and the fuel storage area within one (1) hour prior to the start of fuel loading.

2.14 The overload/Underload setpoints for Refueling Machine and Spent Fuel Handling Machine are adjusted to wet and/or operating conditions.

2.15 Continuous area radiation monitoring will be provided during fuel handling and fuel loading operations.

3.0 TEST METHOD

3.1 Fuel assemblies and neutron sources are inserted in the reactor vessel in accordance with the pre-specified and approved loading sequence.

3.2 Verify the fuel assembly serial numbers, orientations, and locations including the location of neutron sources using underwater TV camera.

3.3 Perform the final review of all hoist position data after all fuel is loaded into the core.

3.4 Perform the final review of the axial centerlines of the fuel assembly upper end fittings in all row locations.

3.5 Maintain a display for indicating the status of the core and spent fuel pool, as well as appropriate records of core loading.

3.6 Collect representative boron sample from the in–service SCS loop at one (1) hour intervals within 8 hours of commencement of initial fuel loading.

3.7 Cease fuel loading and initiate emergency boration, if RCS boron concentration has been possible dilution as indicated by unexplained increase in water level in the reactor vessel.

3.8 Maintain constant communication between fuel handling personnel and the control room within one hour prior to the start of fuel loading.

3.9 ICRR is plotted through the fuel loading using temporary and permanent startup detectors to predict the possibility of approach to criticality. New base count rate data is determined after neutron source or temporary startup detector is moved.

4.0 DATA REQUIRED

4.1 The Z-axis elevation of the dummy fuel assembly during final indexing.

4.2 The as-built bridge and trolley coordinates for each core location.

4.3 As-built core load map.

4.4 Signals of two temporary fuel loading channels and two permanent startup channels.

5.0 ACCEPTANCE CRITERIA

5.1 The serial numbers, orientations, and locations of all fuel assemblies and the locations of the neutron sources are visually verified following completion of core loading.

5.2 All fuel assemblies loaded are in allowable range of the Z-axis elevation (hoist readings on the refueling machine) measured during final indexing.

5.3 A check of the centerlines of all fuel assemblies is confirmed them to be aligned in allowable range of their as-built positions, thereby permitting proper engagement of the upper end fitting with the Upper Guide Structure (UGS).

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

5.4 Boron Dilution Alarm System(BDAS) operate as designed

Initial Criticality and Low-Power Physics Test

14.2.12.3 ~~Low-Power Physics Test~~

Added Next pages as Initial Criticality

~~14.2.12.3.1~~ Low-Power Biological Shield Survey Test

14.2.12.3.2

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

14.2.12.3.1 Initial Criticality Test

1.0 OBJECTIVE

1.1 To attain initial criticality with a safe, organized method.

1.2 To verify that a one decade (or larger) overlap exists between the ex-core startup channels and the ex-core safety log power channels.

2.0 PREREQUISITES

2.1 All CEAs are inserted.

2.2 The RCS boron concentration is sufficiently high.

2.3 The measuring devices for Inverse Count Rate Ratio (ICRR) are set up and operable.

2.4 The reactor coolant system temperature and pressure are stable at the hot zero power.

3.0 TEST METHOD

3.1 Perform initial criticality with boron dilution. CEAs are withdrawn before the boron dilution begins.

3.2 At specific hold points during CEA withdrawal and boron dilution, collect plant data to plot ICRR which is used to predict the critical point.

3.3 When criticality is achieved, verify that one decade overlap exists between the startup channels and the safety log power channels.

4.0 DATA REQUIRED

4.1 Excore instrumentation Startup channels signal and Safety log power channels signal.

4.2 RCS and Pressurizer boron concentration

4.3 CEA Position

4.4 RCS pressure and temperature

5.0 ACCEPTANCE CRITERIA

5.1 The initial criticality is obtained with safe and predicted manner.

5.2 1/2 counts per seconds is registered on startup channels before the effective multiplication factor (k-eff) is 0.98.

5.3 A minimum of one decade overlap exists between the startup channels and the safety log power channels.

4.4.5 Testing and Verification

4.4.5.1 RCS Flow Measurement

RCS flow measurement tests are performed as part of the startup program. The tests verify that the measured RCS flow exceeds the design minimum flow rates used in the safety analysis, but is less than the design maximum flow rate. Also, the tests verify that the measured RCS flow coastdown is conservative with respect to the coastdown used in the safety analysis. The test program is described in Subsection ~~14.2.12.2.3~~. 14.2.12.2.4

4.4.5.2 Component and Fuel Inspections

Inspections performed on manufactured fuel are described in Subsection 4.2.4. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering factors in the design analyses (Subsection 4.4.2.2.3) are met.

4.4.6 Instrumentation Requirements

4.4.6.1 Thermal Power

The DNBR-related core condition and linear heat rate are limited by core protection calculator system, which is part of reactor protection system. The system is described in Section 7.2.

Various reactor trip functions are provided to limit core power and adverse thermal-hydraulic conditions. These trips are variable overpower trip, high logarithmic power level trip, high local power density trip, and low departure from nucleate boiling ratio trip.

To determine the core thermal-hydraulic condition, CEA position, ex-core neutron flux, and reactor coolant flow are measured. These measurements are addressed in Subsection 7.2.1.

4.4.6.2 Power Distribution

There are 61 in-core instrumentation (ICI) assemblies with five self-powered rhodium detectors in each assembly. The ICI assemblies are strategically distributed about the

14.2.7.1.10 Reference Appendix A, Section 5.mm

This section requires that the dynamic response of the plant to automatic closure of all main steam isolation valves (MSIVs) be demonstrated from full power. Performance of this test could result in the opening of main steam atmospheric dump valves at automatic control mode and primary and secondary safety valves. Instead, the dynamic response of the plant can be obtained during the performance of the turbine trip test when the turbine stop valves are closed. The turbine trip test from full power results in essentially similar dynamic plant response and should provide reasonable assurance that primary and secondary safety valves do not lift open during the test. For these reasons, the plant response to automatic closure of all MSIVs from full power is not demonstrated.

14.2.7.1.11 Reference Appendix C, Section 3

This section requires that a neutron count rate of at least 0.5 count per second be registered on the startup channels before the startup begins. The design criterion calls for a neutron count rate of 0.5 count per second with all CEAs fully withdrawn and a multiplication of 0.98. Therefore, prior to the initiation of the initial approach to criticality, the startup channels may record significantly less than 0.5 count per second, but prior to exceeding a multiplication of 0.98, the desired neutron count rate of 0.5 count per second is achieved.

14.2.7.1.12 Reference Appendix C, Section 4

The standard test plateau power levels of 20, 50, 80, and 100 percent are used instead of the recommended power levels of 25, 50, 75, and 100 percent.

14.2.7.1.13 Reference Section C, Regulatory Position 4

14.2.12.2.2

This section requires inclusion of acceptance criteria that account for uncertainties. The test summaries in Subsections ~~14.2.12.2.1~~ and 14.2.12.1.46 are essential to the demonstration of conformance to the requirements for structures, components, and features important to safety.

2.0 PREREQUISITES

2.1 Test instrumentation is available and calibrated.

2.2 Construction activities on the RCS and associated systems are completed.

2.3 All permanently installed instrumentation on the system to be tested is available and calibrated.

3.0 TEST METHOD

3.1 Determine the RCS heat loss using the steam-down method:

3.1.1 Stabilize the steam generator levels with the RCS at HZP conditions.

3.1.2 Secure steam generator feedwater and blowdown.

3.1.3 Measure both the pressurizer heater power required to maintain RCS temperature and pressure and RCP power.

3.1.4 Perform a heat balance calculation to determine heat loss.

3.2 Determine the pressurizer heat loss, with(1) and without continuous spray flow, by measuring the pressurizer heater power required to maintain the RCS at HZP conditions, and then performing a heat balance calculation.

4.0 DATA REQUIRED

4.1 RCS temperatures

14.2.12.2.7

(1) Pressurizer heat loss with continuous spray flow to be determined during post-core hot functional test after spray valve adjustments have been performed per Subsection ~~14.2.12.2.6~~.

4.0 DATA REQUIRED

4.1 As specified by the individual post-core hot functional test ~~appendices~~

5.0 ACCEPTANCE CRITERIA

5.1 Integrated operation of the primary, secondary, and related auxiliary systems is in accordance with the system descriptions.

~~Loose Parts~~5.2 All HFTs have been performed and have met their respective acceptance criteria.

5.3 The integrated operation is demonstrated by the successful heat up and pressurization of the plant from cold shutdown to hot standby.

~~14.2.12.2.2~~ NSSS Integrity Monitoring System (Post-core)

14.2.12.2.3

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To obtain baseline data ~~on the loose parts monitoring system (~~for Acoustic Leak Monitoring system (ALMS), Loose Parts Monitoring system (LPMS) and RCP Vibration Monitoring system (RCPVMS) during post-core HFT.

1.2 To ~~adjust LPMS~~verify existing, or establish new alarm setpoints as ~~necessary~~required for the NSSS Integrity Monitoring System.

2.0 PREREQUISITES

2.1 ~~Preoperational tests of~~Plant is stable at the ~~LPMS have been completed.~~

~~2.2 All LPMS instrumentation has been calibrated~~required temperature and ~~is operable~~pressure plateau.

2.2 The NIMS is operational as applicable.

3.0 TEST METHOD

3.1 Collect baseline data ~~using the LPMS~~ during ~~plant heatup and at normal operating conditions~~post-core HFT.

3.2 ~~Analyze baseline data and, if necessary, adjust alarm~~Adjust setpoints, if required, based on the data collected.

4.0 DATA REQUIRED

4.0 DATA REQUIRED

4.1 Baseline data ~~using~~for ALMS, LPMS and RCPVMS

4.2 ~~LPMS alarm~~Alarm setpoints for ALMS, LPMS and RCPVMS

4.3 RCS temperature and pressure

5.0 ACCEPTANCE CRITERIA

5.1 Baseline data of ALMS, LPMS ~~performs as described in Subsection 7.7.1.5~~and RCPVMS are collected during post-core HFT.

5.2 The ~~LPMS~~ alarm setpoints ~~have been~~of each subsystem are adjusted as necessary.

~~14.2.12.2.3~~ Post-Core Reactor Coolant System Flow Measurements

14.2.12.2.4

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To determine the post-core reactor coolant system (RCS) flow rate and flow coastdown characteristics

1.2 To establish reference post-core RCS pressure drops

~~14.2.12.2.4~~ Post-Core Control Element Drive Mechanism Performance

14.2.12.2.5

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate the proper operation of the control element drive mechanisms (CEDMs) and control element assemblies (CEAs) under hot shutdown and hot zero-power (HZP) conditions

1.2 To verify proper operation of the CEA position indicating system and alarms

1.3 To measure CEA drop times

2.0 PREREQUISITES

2.1 The CEDM control system (CEDMCS) pre-core performance test has been completed.

2.2 All test instrumentation is available and has been calibrated.

2.3 The information processing system is operational.

2.4 The CEDM cooling system is operational.

2.5 CEDM coil resistance has been measured.

3.0 TEST METHOD

3.1 Perform the following at hot shutdown conditions:

3.1.1 Withdraw and insert each CEA to verify proper operation of CEDMs.

3.2 Perform the following at HZP conditions:

5.5 IPS major/minor deviation alarm operates as designed.

5.6 CEA Deviation alarm operate as designed

~~14.2.12.2.5~~ Post-Core Reactor Coolant and Secondary Water Chemistry Data

14.2.12.2.6

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.

3.2 Perform RCS and ~~steam generator~~secondary system sampling and chemistry analysis after every significant change in plant conditions (i.e., heatup, cooldown, chemical additions).

4.0 DATA REQUIRED

4.1 Plant conditions

4.2 ~~Steam generator~~Secondary system chemistry analysis

4.3 RCS chemistry analysis

5.0 ACCEPTANCE CRITERIA

5.1 RCS and ~~steam generator~~secondary system water chemistry ~~can be~~are maintained within design limits as ~~described~~specified in ~~Subsections~~subsection 9.3.4 and 10.3.5.

5.2 Baseline data for the ~~steam generators~~ RCS and ~~RCS~~ secondary system is established.

~~14.2.12.2.6~~ Post-Core Pressurizer Spray Valve and Control Adjustments

14.2.12.2.7

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To establish the proper settings of continuous spray valves

1.2 To measure the rate at which the pressurizer pressure can be reduced using pressurizer spray

2.0 PREREQUISITES

2.1 The RCS is ~~being~~ operated at nominal HZP conditions.

2.2 All permanently installed instrumentation is available and calibrated.

~~14.2.12.2.7~~ Post-Core Reactor Coolant System Leak Rate Measurement

14.2.12.2.8

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To measure the post-core ~~load~~ RCS leakage at HZP conditions

2.0 PREREQUISITES

2.1 Hydrostatic testing of the RCS and associated systems has been completed.

2.2 The RCS and chemical and volume control system (CVCS) are operating as a closed system.

2.3 The RCS is at HZP conditions.

2.4 All permanently mounted instrumentation is properly calibrated.

3.0 TEST METHOD

3.1 Measure and record the changes in water inventory of the RCS and CVCS for a specified interval of time.

4.0 DATA REQUIRED

4.1 Pressurizer pressure, level, and temperature

4.2 Volume control tank level, temperature, and pressure

4.3 Reactor drain tank level, temperature, and pressure

4.4 RCS temperature and pressure

4.5 Safety injection tank level and pressure

4.6 Time interval

5.0 ACCEPTANCE CRITERIA

5.1 Identified and unidentified leakage are within the limits described in the Technical Specifications and as described in Subsection 5.2.5.

~~14.2.12.2.8~~ Post-Core In-Core Neutron Flux Detector Test

14.2.12.2.9

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To measure the leakage resistance of the fixed in-core detectors

1.2 To verify the proper insulation of the entire Fixed In-core Detector Amplifier System(FIDAS) signal cable network including detectors, mineral insulated cables, and organic cables to the FIDAS Cabinet.

1.3 To verify the proper operation of the FIDAS

2.0 PREREQUISITES

2.1 All permanently installed in-core neutron flux detectors are properly calibrated.

2.2 Installation and preoperational checkout of the in-core neutron flux detectors are completed.

2.3 Special test equipment is available and calibrated.

3.0 TEST METHOD

3.1 Measure and record the leakage resistance of each in-core detector at the nominal HZP condition.

4.0 DATA REQUIRED

4.1 Resistance measurements

5.0 ACCEPTANCE CRITERIA

5.1 Insulation resistance of the in-core neutron flux detectors is as described in manufacturer's recommendations.

5.2 Each of test item listed below shall be satisfied.

- Rhodium detector raw signal quality
- Background detector raw signal quality
- Rhodium detector charge quality
- Rhodium detector sensitivity quality
- Compensated/uncompensated neutron flux quality

~~14.2.12.2.9~~ Post-Core Instrument Correlation

14.2.12.2.10

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate the proper operation of the plant protection system (PPS), core protection calculators (CPCs), information processing system (IPS), and qualified indication and alarm system (QIAS).

2.0 PREREQUISITES

2.1 PPS, CPCs are in operation.

2.2 IPS, QIAS, and core operating limit supervisory system (COLSS) are in operation.

- Steam generator pressure
- Steam generator primary side differential pressure
- Reactor vessel differential pressure
- Containment pressure

~~14.2.12.2.10~~ Post-Core Acoustic Leak Monitoring System

14.2.12.2.11

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To obtain baseline data on the acoustic leak monitoring system (ALMS) at various power level

1.2 To adjust ALMS alarm setpoints as necessary

1.3 To obtain the ALMS sensor baseline data

1.4 To verify the crack detection function at various power level

2.0 PREREQUISITES

2.1 Preoperational tests on ALMS have been completed.

2.2 All ALMS instrumentation has been calibrated and is operable.

3.0 TEST METHOD

3.1 Collect baseline data using the ALMS during plant heatup and at normal operation conditions.

4.0 DATA REQUIRED

4.1 Baseline data using ALMS

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.

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

14.2.12.2.12

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.

5.0 ACCEPTANCE CRITERIA

5.1 ~~Baseline neutron and gamma surveys have been completed.~~

~~5.2~~ The biological shield survey test performs as described in Subsection 12.3.2.2.

5.2 Radiation levels shall be less than the maximum specified for the applicable zone.

5.3 Accessible areas and occupancy time during power operation shall be within the design values.

~~14.2.12.3.2~~ Isothermal Temperature Coefficient Test

14.2.12.3.3

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To measure the ~~isothermal temperature coefficients (ITCs)~~ITC for various ~~reactor coolant system (RCS) temperatures, pressures, and control element assembly (~~CEA~~)~~ configurations

1.2 To determine the ~~moderator temperature coefficient (~~MTC~~)~~ from the measured ITC

2.0 PREREQUISITES

2.1 The reactor is critical with a stable boron concentration and the desired CEA configuration and RCS temperature and pressure.

2.2 The reactivity computer is operable.

3.0 TEST METHOD

3.1 Changes in RCS temperature are introduced and the resultant changes in reactivity measured.

~~14.2.12.3.3~~ Shutdown and Regulating Control Element Assembly Group Worth Test

14.2.12.3.4

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To determine regulating and shutdown CEA group worths necessary to demonstrate ~~adequate~~ shutdown margin

1.2 To demonstrate that the shutdown margin is adequate

2.0 PREREQUISITES

2.1 The reactor is critical.

2.2 The reactivity computer is operating.

3.0 TEST METHOD

3.1 Measurement of regulating and shutdown CEA groups:

3.1.1 The CEA group worths are measured either by the dilution/boration of the RCS or by using the CEA exchange method.

3.1.2 Worths may be determined by the CEA drop ~~and/or by use of alternate CEA configurations~~or insertion.

4.0 DATA REQUIRED

4.1 ~~Conditions of the measurement:~~

~~4.1.1 RCS temperature~~

~~4.1.2 Pressurizer pressure~~

~~4.1.3 CEA configuration~~

~~4.1.4 Boron concentration~~

~~4.2 Time-dependent information:~~

~~4.2.1~~ Reactivity variation depending on CEA movement

~~4.2.2 CEA positions~~

4.2 CEA group worths and total CEA worth are calculated with the reactivity data.

5.0 ACCEPTANCE CRITERIA

5.1 The measured CEA group worths ~~agree with predictions~~shall be within the ~~acceptance criteria specified in Table 14.2-6.~~design values of the predicted worths.

5.2 ~~Evaluation of the measurements verifies shutdown margin.~~ The measured total CEA worth shall be within the design values of the predicted worth.

~~14.2.12.3.4~~ Differential Boron Worth Test

14.2.12.3.5

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To measure the differential boron reactivity worth for various CEA configurations

2.0 PREREQUISITES

2.1 CEA group worth tests are completed.

2.2 Critical configuration boron concentration tests are completed.

3.0 TEST METHOD

3.1 The differential boron worths are determined from the measured boron concentrations associated with the state points measured during the CEA group worth tests.

4.0 DATA REQUIRED

4.1 ~~Conditions of the measurement at state points:~~

~~4.1.1 RCS temperature~~

~~4.1.2 Pressurizer pressure~~

~~4.1.3 CEA configuration~~

~~4.1.4~~ Boron concentration change between state points

4.2 Integral reactivity changes between state points

5.0 ACCEPTANCE CRITERIA

5.1 The measured boron worths ~~agree with~~shall be within the design values of the predicted values ~~within the acceptance criteria specified in Table 14.2-6.~~ .

~~14.2.12.3.5~~ Critical Boron Concentration Test

14.2.12.3.6

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To measure the critical boron ~~concentrations~~concentration for various CEA configurations at appropriate temperatures and associated pressures

~~14.2.12.3.6~~ Control Element Assembly Symmetry Test

14.2.12.3.7

1.0 ~~OBJECTIVE~~OBJECTIVES

1.1 To demonstrate that no loading or fabrication errors that result in measurable CEA worth asymmetries have occurred~~. This objective can be satisfied by performing CEA group worth measurement and by performing low-power (≤ 20 percent power) distribution measurements.~~

2.0 PREREQUISITES

2.1 The reactivity computer is in operation.

2.2 The reactor is critical at the desired conditions~~ with the controlling CEA group partially inserted and in manual control~~.

3.0 TEST METHOD

3.1 Conduct CEA symmetry test~~ (at hot zero-power [HZP] conditions)~~.

3.1.1 Insert the first CEA of a symmetric group with all remaining CEAs withdrawn except the controlling group, which is positioned for zero reactivity.

3.1.2 Withdraw the inserted CEA while another CEA in the symmetric group is inserted and determine the differences in worth (net reactivity) of the CEAs from the reactivity computer.

3.1.3 Sequentially swap the remainder of the CEAs~~ the symmetric group has been determined~~.

3.1.4 Repeat steps 3.1.1, 3.1.2, and 3.1.3 for the remainder of the symmetric groups.

Initial Fuel Loading and Post-Core Hot Function Tests

LIST OF TABLES

Initial Criticality and Low-Power Physics Test

14.2.12.2.1 Initial Fuel Loading

Initial Fuel Loading and Post-Core Hot Function Tests

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

14.2.12.2.2
14.2.12.2.3
14.2.12.2.4
14.2.12.2.5
14.2.12.2.6
14.2.12.2.7
14.2.12.2.8
14.2.12.2.9
14.2.12.2.10
14.2.12.2.11
14.2.12.2.12

Initial Criticality and Low-Power Physics Test

14.2.12.3.1 Initial Criticality

Table 14.2-3

~~Low-Power Physics Tests~~

Subsection	Test
~~14.2.12.3.1~~	Low-power biological shield survey test
~~14.2.12.3.2~~	Isothermal temperature coefficient test
~~14.2.12.3.3~~	Shutdown and regulating control element assembly group worth test
~~14.2.12.3.4~~	Differential boron worth test
~~14.2.12.3.5~~	Critical boron concentration test
~~14.2.12.3.6~~	Control element assembly symmetry

14.2.12.3.2
14.2.12.3.3
14.2.12.3.4
14.2.12.3.5
14.2.12.3.6
14.2.12.3.7

14.2.12.3.1

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	~~14.2.12.3.3~~ ~~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	~~14.2.12.3.3~~	Shutdown and regulating CEA group worth test
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 14.2.12.4.17 14.2.12.4.9	Steady-state core performance test Core operating limit supervisory system verification 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

14.2.12.2.5

14.2.12.2.5
14.2.12.2.8
14.2.12.2.2
14.2.12.2.6
14.2.12.2.3
14.2.12.2.4
14.2.12.2.11
14.2.12.2.12
14.2.12.2.9

14.2.12.3.3
14.2.12.3.6
14.2.12.3.4
14.2.12.2.10
14.2.12.2.12
14.2.12.2.10
14.2.12.2.12

Initial Fuel Loading

Initial Criticality

Table 14.2-7 (14 of 18)

RG 1.68 APP. A	Subsection #	Individual Test
4.i	14.2.12.1.27 14.2.12.1.54 ~~14.2.12.2.4~~	Digital rod control system test Pre-core control element drive mechanism performance test Post-core control element drive mechanism performance
4.j	14.2.12.4.21	HVAC capability test
4.k	14.2.12.1.34 14.2.12.1.62	Auxiliary feedwater system test Main turbine systems test
4.l	14.2.12.1.64	Main steam isolation valves and ~~MSIV bypass valves~~MSIVBVs test
4.m	-	Not applicable This is not a design feature of the APR1400.
4.n	14.2.12.4.3 14.2.12.4.11	Control systems checkout test Intercomparison of plant protection system, core protection calculator, information processing system, and qualified information and alarm system inputs
4.o	~~14.2.12.2.4~~	Post-core control element drive mechanism performance
4.p	14.2.12.1.3 14.2.12.1.63	Pressurizer pilot-operated safety relief valve test Main steam safety valve test
4.q	14.2.12.1.5 14.2.12.1.7	Chemical and volume control system letdown subsystem test Chemical and volume control system charging subsystem test
4.r	14.2.12.1.20 14.2.12.1.65	Shutdown cooling system test Main steam system test
4.s	-	Exception Reactor internal vibration test is excluded from the comprehensive vibration assessment program described in Subsection 3.9.2.4 since APR1400 is classified as non-prototype category I plant according to NRC RG 1.20 (Reference 9).
4.t	14.2.12.4.22	Natural circulation test
4.u	14.2.12.4.3	Control systems checkout test
5.a	14.2.12.4.1	Variable Tavg (isothermal temperature coefficient and power coefficient) test
5.b	14.2.12.4.10 14.2.12.4.12	Steady-state core performance test Verification of core protection calculator power distribution related constants test

14.2.12.2.5

14.2.12.2.5

Table 14.2-7 (15 of 18)

RG 1.68 APP. A	Subsection #	Individual Test
5.c	-	Not applicable This is not a design feature of the APR1400.
5.d	14.2.12.4.2 14.2.12.4.14 14.2.12.4.17	Unit load transient test Core protection calculator verification Core operating limit supervisory system verification
5.e	14.2.12.4.26	Pseudo dropped CEA test
5.f	14.2.12.1.27 ~~14.2.12.2.4~~	Digital rod control system test Post-core control element drive mechanism performance
5.g	~~14.2.12.2.4~~	Post-core control element drive mechanism performance
5.h	-	Not applicable This is not a design feature of the APR1400.
5.i	14.2.12.4.2 14.2.12.4.6	Unit load transient test Unit load rejection test
5.j	14.2.12.1.21 14.2.12.1.22 14.2.12.1.59	Safety injection system test Safety injection tank subsystem test Pre-core safety injection check valve test
5.k	14.2.12.4.7 14.2.12.4.13 14.2.12.4.15	Shutdown from outside the main control room test Feedwater and auxiliary feedwater system test Main steam atmospheric dump and turbine bypass valve test
5.l	~~14.2.12.2.3~~	Reactor coolant system flow measurements
5.m	14.2.12.4.18	~~Baseline nuclear steam supply system~~NSSS integrity monitoring system
5.n	~~14.2.12.2.10~~	Post-core acoustic leak monitor system test
5.o	-	Exception Reactor internal vibration test is excluded from the comprehensive vibration assessment program described in Subsection 3.9.2.4 since the APR1400 is classified as non-prototype category I plant according to NRC RG 1.20 (Reference 9).
5.p	-	Not applicable This is not a design feature of the APR1400.
5.q	14.2.12.4.3 14.2.12.4.14 14.2.12.4.17	Control systems checkout test Core protection calculator verification Core operating limit supervisory system verification

14.2.12.2.5

14.2.12.2.5

14.2.12.2.4

14.2.12.2.11