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SUBJECT: Forwards rev to FSAR Table 14.2-206, "CFR Sys Startup Test 5 (SUT-5)," superseding 860530 version & incorporating NRC comments. Rev will be included in future FSAR amend.

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August 15, 1986
(NMP2L 0817)

Ms. Elinor G. Adensam, Director
BWR Project Directorate No. 3
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Washington, DC 20555

Dear Ms. Adensam:

Re: Nine Mile Point Unit 2
Docket No. 50-410

This letter submits a revision to Nine Mile Point Unit 2 (NMP2) Final Safety Analysis Report (FSAR) Table 14.2-206, Control Rod Drive System Startup Test No. 5 (SUT-5). This revision (shown in Attachment 1) will be included in a subsequent amendment of the Final Safety Analysis Report. This revision supersedes the version transmitted by letter NMP2L-0724 dated May 30, 1986 and incorporates the concerns and comments of NRC Region 1 and NRR reviewers.

Included in Attachment 2 are updated pages which incorporate changes not previously submitted to the Nuclear Regulatory Commission. Most of the changes on these pages incorporate Nuclear Regulatory Commission's comments and editorial comments of our staff.

With this submittal, it is Niagara Mohawk's understanding that the NMP2 Accelerated Power Ascension Test Program submitted by letter NMP2L-0724 as amended by this letter and NMP2L-0781 dated July 11, 1986 can be approved by the NRC. Your prompt approval of this program will be appreciated.

Very truly yours,

C V Mangan

C. V. Mangan
Senior Vice President

DLH/LSL:ar
1918G

Enclosures

xc: W. A. Cook, NRC Resident Inspector
Project File (2)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)

Niagara Mohawk Power Corporation) Docket No. 50-410

(Nine Mile Point Unit 2))

AFFIDAVIT

C. V. Mangan, being duly sworn, states that he is Senior Vice President of Niagara Mohawk Power Corporation; that he is authorized on the part of said Corporation to sign and file with the Nuclear Regulatory Commission the documents attached hereto; and that all such documents are true and correct to the best of his knowledge, information and belief.

CVMangan

Subscribed and sworn to before me, a Notary Public in and for the State of New York and County of Onondaga, this 15th day of August, 1986.

Christine Austin
Notary Public in and for
Onondaga County, New York

My Commission expires:
CHRISTINE AUSTIN
Notary Public in the State of New York
Qualified in Onondaga Co. No. 4787687
My Commission Expires March 30, 1987

ATTACHMENT 1

TABLE 14.2-206

CONTROL ROD DRIVE SYSTEM

Startup Test (SUT-5)

Test Objectives

1. To demonstrate that the CRD system operates properly over the full range of primary coolant temperatures and pressures from ambient to operating.

2. To determine the initial operating characteristics of the entire CRD system.

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and initiation of testing. The CRD manual control system preoperational testing must be completed on CRDs being tested. The reactor vessel, closed loop cooling water system, condensate supply system, and instrument air system must be operational to the extent required to conduct the test.

Test Procedure

The CRD tests performed during the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all CRDs operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all CRD tests to be performed during startup testing is as follows:

CONTROL ROD DRIVE SYSTEM TESTS

		Test Conditions Reactor Pressure with Core Loaded psig (kg/cm^2)			
Action	Accumulator Pressure	0	600(42.2)	800(56.2)	Rated
Position Indication		All			

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TABLE 14.2-206 (Cont)

CONTROL ROD DRIVE SYSTEM TESTS

Action	Accumulator Pressure	Test Conditions Reactor Pressure with Core Loaded psig (kg/cm²) 0	600(42.2)	800(56.2)	Rated
Normal Stroke Times Insert/ Withdraw		All			4*
Coupling		All***			
Friction		All			4*
Scram	Normal	All	4*	4*	All
Scram	Minimum	4*			
Scram	Zero				4*
Scram	Normal				4**

* Refers to four CRDs selected for continuous monitoring based on slow normal accumulator pressure scram times or unusual operating characteristics, at zero reactor pressure or rated reactor pressure when this data is available. The four selected CRDs must be compatible with the rod worth minimizer, RSCS system, and CRD sequence requirements.

** Scram times of the four slowest CRDs (based on scram data at rated pressure) will be determined at test conditions 2 and 6 during planned reactor scrams.

*** Establish that this check is normal operating procedure.

NOTE: Single CRD scrams should be performed with the charging valve closed. (Do not ride the charging pump head.)

TABLE 14.2-206 (Cont)

Criteria

Level 1:

a) Each CRD must have a normal withdraw speed less than or equal to 3.6 inches per second (9.14 cm/sec), indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

b) The mean scram time of all operable CRDs must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized)

Position Inserted from Fully Withdrawn	Scram Time (Seconds)
45	0.43
39	0.86
25	1.93
05	3.49

c) The mean scram time of the three fastest CRDs in a two-by-two array must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

Position Inserted from Fully Withdrawn	Scram Time (Seconds)
45	0.45
39	0.92
25	2.05
05	3.70

d) The maximum scram insertion time of each control rod from the fully withdrawn position to notch position 5, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

Level 2:

a) Each CRD must have a normal insert or withdraw speed of 3.0 ± 0.6 ips (7.62 ± 1.52 cm/sec), indicated by a full 12-ft stroke in 40 to 60 sec.

TABLE 14.2-206 (Cont)

b. With respect to the CRD friction tests, if the differential pressure variation exceeds 15 psid (1.1 kg/cm^2) for a continuous drive in, a settling test must be performed, in which case the differential settling pressure should not be less than 30 psid (2.1 kg/cm^2) nor should it vary by more than 10 psid (0.7 kg/cm^2) over a full stroke.

ATTACHMENT 2

3. Pump head, capacity.

4. System flows.

5. Heat-up characteristics, when attainable.

6. Tuning of system controls.

7. Response to simulated safety signals and/or loss of power.

8. Operating times.

This phase of testing verifies the ability of the plant to support fuel load and power operations.

14.2.1.4 Initial Startup Test Phase

The initial startup test phase commences with the receipt of the operating license and the preparation for fuel load and extends through the 100-percent rated power/100-hr warranty demonstration. The initial startup test phase is divided into ~~six~~ seven testing plateaus: open vessel (including fuel loading), heatup, test plateaus 1,2,3,4, and warranty run. ~~rated power warranty run.~~ Testing performed during this phase of the program ensures that fuel loading is accomplished in a safe manner, confirms the plant design basis, demonstrates, to the extent possible, the plant's ability to withstand anticipated transients and postulated accidents and verifies that the plant can be safely brought to rated power and sustained power operations.

14.2.2 Organization and Staffing

The Unit 2 startup and test organization and interfaces to plant operations, SWEC, General Electric, and other selected NMPC organizations are shown in Figure 14.2-7 and are discussed in the following sections.

The Unit 2 operational organization is discussed in
Chapter 13.0. The initial startup test phase is performed
under the control of the station superintendent and
coordinated by the Power Ascension Manager. The 27
responsibilities of the operations organization during the
startup and test program are discussed in the following
sections.

Staffing levels during the startup and test program will be commensurate with schedule and project needs and requirements.

14.2.10.1.5 Initial Fuel Loading (Open Vessel Plateau)

Fuel loading requires the movement of the full core complement of assemblies from the fuel pool to the core, with each assembly identified by number before being placed in the correct coordinate position. The procedure controlling this movement is arranged so that operability checks of installed neutron instrumentation are made at predetermined intervals throughout the loading, thus demonstrating reliable monitoring capability to ensure subcriticality is maintained throughout fuel loading. A complete check is made of the fully loaded core to ascertain that all assemblies are properly installed, correctly oriented, and occupying their designated positions.

14.2.10.1.6 Zero Power Level Tests (Open Vessel Plateau)

At this point, a number of tests are conducted that are best described as initial zero power level tests. Chemical and radiochemical tests are made in order to check the quality of the reactor water before and after fuel loading and to establish base and background levels that are required to facilitate later analysis and instrument calibrations. Plant and site radiation surveys are made at specific locations for comparison with the values obtained at the subsequent operating power levels. Control rod drive system testing takes place ~~while the reactor vessel is assembled~~ in 27
preparation for initial criticality and initial heatup.

14.2.10.2 Initial Criticality and Heatup to Rated Temperature and Pressure 27

Initial criticality and heatup follows the satisfactory 27
completion of the fuel loading and zero power level tests (Sections 14.2.10.1.5 and 14.2.10.1.6). Further checks are 27
made of coolant chemistry together with radiation surveys at the selected plant locations. All CRDs are scram-timed ~~at rated pressure~~ prior to initial heatup. The process 27
computer checkout continues as more process variables become available for input. The reactor core isolation cooling (RCIC) system will undergo controlled starts at low reactor pressure and at rated conditions, with testing in the quick-start mode at rated pressure. Correlations are obtained 27
between reactor vessel temperatures at several locations and the values of other process variables as heatup continues. The movements of NSSS piping in the drywell, mainly as a function of expansion are recorded for comparison with design data.

selected flow conditions; the work involves determination of the core thermal power, maximum linear heat generation rate, minimum critical power ratio (MCPR), and other thermal parameters.

11. Overall plant stability in relation to minor perturbations is shown by the following group of tests that are made at ~~all~~ appropriate test points:

~~a. Core power-void mode response.~~

~~b.~~ a. Pressure regulator set point change.

~~c.~~ b. Water level setpoint change.

~~d.~~ c. Bypass valve opening.

~~e.~~ d. Recirculation flow setpoint change.

~~For the first of these tests, neutron flux (power) response on LPRM chambers is observed on control rod withdrawal.~~ The ~~next~~ first two tests require that the changes made approximate as closely as possible to a step change in demand, while for the next test the bypass valve is opened quickly. The remaining test is performed to properly adjust the control loop of the recirculation system. For all of these tests, plant performance is monitored by recording the transient behavior of numerous process variables, the one of principal interest being neutron flux. Other imposed transients are produced by step changes in demand core flow, simulating loss of a feedwater heater and failure of the operating pressure regulator to permit takeover by the backup regulator. 22

12. The category of major plant transients includes full closure of all MSIVs, fast closure of turbine generator control valves and stop valves, loss of the main generator and offsite power, tripping of a feedwater pump, and ~~several~~ trips of the recirculation pumps. The plant transient behavior is recorded for each test and the results may be compared with the acceptance criteria and the predicted design performance.

13. A test is made of the main steam safety relief valves in which leaktightness and general operability are demonstrated.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

Not originally part of LDCN 586

14.2.12.2 General Discussion of Initial Startup Tests

All tests comprising the initial startup test phase are discussed in Tables 14.2-201 through 14.2-~~306.~~307 A test objective, test prerequisites, test description, and statement of test acceptance criteria are provided for each test where applicable.

The operating and safety-oriented characteristics of the plant being explored are described in the test objectives.

Where applicable, a definition of the relevant acceptance criteria for the test is given and designated either Level 1 or Level 2. A Level 1 criterion normally relates to the value of a process variable assigned in the design of the plant, component system or associated equipment. If a Level 1 criterion is not satisfied, the plant is placed in a stable condition until resolution is obtained. Tests compatible with this stable condition may be continued. Following resolution, applicable tests are repeated as necessary to verify that the requirements of the Level 1 criterion are now satisfied.

A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be started. 22

For transients involving oscillatory response, the criteria are specified in terms of decay ratio (defined as the ratio of successive maximum amplitudes of the same polarity). The decay ratio must be less than unity to meet a Level 1 criterion and less than 0.25 to meet Level 2.

During the conduct of the initial startup test phase the technical specifications will be followed.

TABLE 14.2-207 (Cont)

Acceptance Criteria

Level 1:

1. There is a neutron signal-to-noise count ratio of at least 2 to 1 on the required operable SRMs or fuel loading chambers.

22 2. Minimum count rate is in accordance with the technical specifications.

Level 2:

Not applicable.

3. Each IRM channel must be on scale before the SRMs exceed their rod block setpoint.

TABLE 14.2-244 (Cont)

Action	Test Conditions
1. Controller adjustment based on subsystem perturbations	a. Reactor not isolated above 10% rated power but ≤25% rated power. b. RHR system in steam condensing mode. c. RCIC flow to CST.
2. Demonstration of steam condensing mode.	a. Reactor at hot standby and isolated. b. RCIC flow to RPV.
3. Take heat exchanger capacity data.	a. RHR in shutdown cooling mode. b. 27 After trip or cooldown from TC-6 or during the first shutdown after the test program in order to provide sufficient decay heat. c. RHR in suppression pool cooling mode.

Acceptance Criteria

Level 1:

The transient response of any system-related variable to any test input must not diverge.

Level 2:

1. The RHR system must be capable of operating in the steam condensing, suppression pool cooling, and shutdown
27 cooling modes (with ~~both~~ one or both heat exchangers) at heat removal rates equivalent to or greater than the values indicated on the process diagrams.

2. System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

TABLE 14.2-303

BOP PIPING VIBRATION

Startup Test (SUT-77)

Test Objective

1. To verify that steady-state and/or transient piping vibration for the main steam (including relief valve discharge), residual heat removal, feedwater, reactor core isolation cooling, and condensate systems are within acceptable limits.

2. To verify that steady-state vibrations for small bore piping and essential instrumentation lines on main steam, nuclear steam supply, feedwater, reactor plant sampling, residual heat removal, and reactor core isolation cooling are within acceptable limits.

Prerequisites

The appropriate preoperational tests and generic pipe vibration tests have been completed, and the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate.

Test Procedure

Due to the high radiation levels involved no visual observations are performed in this test. Remote monitoring of piping vibration will be utilized. The locations to be monitored and the corresponding predicted displacements will be provided in the startup test procedure.

The following tests are performed:

Action	Test Conditions	
1. Record vibration of RCIC lines.	a. In conjunction with RCIC pump start, rated pump flow, and RCIC pump trip (SUT-14 at HU and 1).	27
2. Record vibration of main steam lines.	a. ~~At approximately 25, 50, 75, and 100% of rated thermal power.~~ At TC-2,3,5, and 6.	

[illegible]

TABLE 14.2-303 (Cont)

Action	Test Conditions
	b. In conjunction with pressure controller setpoint changes (SUT-22 at TC-2 and 4), MSIV closure (SUT-25 at TC-3 and 5), relief valve capacity checks (SUT-26 at TC-1), and turbine generator trip (SUT-27 at TC-2). deleted TC, 4
3. Record vibration of main steam relief valve lines.	a. In conjunction with relief valve capacity checks (SUT-26 at TC-1)
4. Record vibration of feedwater and condensate lines.	a. At ~~approximately 25, 50, 75, and 100% of rated thermal power.~~ TC-2, 3, 5, and 6. b. In conjunction with turbine generator trip (SUT-27 at TC-2 and 6) and feedwater system tests (SUT-23 at TC-1, 2, 5, and 6).
27 5. Record vibration of residual heat removal lines.	~~a. At approximately 25, 50, 75, and 100% of rated thermal power.~~ ~~b.~~ a. In conjunction with RHR steam condensing mode (SUT 71 at TC 1 and 6) and shutdown cooling mode (SUT-71 at ~~TC-1 and 6~~ TC-6).
~~6. Record vibration of reactor plant sampling lines.~~	~~a. In conjunction with feedwater system test (SUT-23 at TC-6).~~
6. ~~7.~~ Record vibration of recirculation system ~~instrumentation.~~ small bore lines	a. ~~In conjunction with rated recirculation flow on 100% load line.~~ At TC-2, 3, 5 and 6.
8. Record vibration of reactor vessel level indicator instrumentation (nuclear boiler instrumentation, ISC) lines.	a. At ~~approximately 25, 50, 75, and 100% of rated thermal power.~~ TC-2, 3, 5, and 6

TABLE 14.2-303 (Cont)

Action	Test Conditions
9. Record vibration of main steam instrumentation lines.	a. In conjunction with MSIV closure (SUT-25 at TC-6).

27

TABLE 14.2-303 (Cont)

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TABLE 14.2-303 (Cont)

Acceptance Criteria[1]

Transient Vibration

Level 1:

The acceptance limits for Level 1 are based upon ASME B&PVC, Section III, Equation 9 for Class 1, 2, and 3 systems or ANSI B31.1, Equation 12 for Class 4 systems. These acceptance limits restrict the bending stress due to deflection plus deadweight and pressure to a value less than the normal/upset allowable stress for occasional loads.

Level 2:

The acceptance limits for Level 2 are based on pipe stress and support loads that do not exceed design basis predictions.

Steady-State Vibration 27

Level:

Acceptance criteria limits are based upon deflection equations given in ANSI/ASME OM-3 with limiting allowable stress of (0.8/1.3) S_{el} for carbon steel piping and an allowable stress of S_a at 10^{11} cycles 27
using curve c for stainless steel piping.

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[1]These criteria are technically equivalent to those described in response to Question F210.37.

TABLE 14.2-302

ESF AREA COOLING

Startup Test (SUT-76)

Test Objective

The purpose of this test is to verify that the unit coolers serving the RCIC, RHR, LPCS, HPCS, SGTS, auxiliary building MCC, service water bay, and standby diesel generator control rooms can maintain the equipment room temperature below the maximum design limits under postulated accident conditions. 27

Prerequisites

The appropriate preoperational tests have been completed. The SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate. The service water system is operational to the extent required to conduct the test.

Test Procedure

The ESF areas listed above will be isolated from the normal ventilation system and major equipment in the area run in the mode providing the maximum practical heat load. Numerous temperature measurements will be made in the area. Adequate temperature and flow data will be collected to perform a heat balance across the area coolers under test conditions. ~~The test data will then be provided to appropriate engineering personnel for evaluation to verify the system will perform adequately under design basis conditions.~~ 27

Acceptance Criteria

Level 1:

All ESF area air space temperatures measured shall not exceed the design limits specified in Table 9.4-1.

Level 2:

Evaluation of test data shall demonstrate that all ESF area air space temperatures will remain below the design limits in Table 9.4-1 under design basis conditions.

TABLE 14.2-304

BOP SYSTEM EXPANSION

Startup Test (SUT-78)

Test Objective

To verify that BOP piping systems are free to expand and move without unplanned obstruction or restraint during system heatup and cooldown cycles and to verify that the associated measured displacements are within specified limits.

Prerequisites

The appropriate preoperational tests have been completed and the SORC has reviewed and approved the test procedures and the initiation of testing. Instrumentation has been checked and calibrated as appropriate.

Test Procedure

Visual inspections will be performed, to the extent 22
possible, to verify freedom of movement. In addition, scribers and remote sensors will be utilized to obtain displacement readings.

Action	Test Conditions
1. Visual inspection for: reactor water cleanup, high pressure core spray, low pressure core spray, residual heat removal, reactor core isolation cooling, and feedwater ~~, and condensate~~ systems.	a. Prior to initial heatup at ambient conditions. b. At the reactor vessel temperature plateau of 300 ± 50°F. c. At the reactor vessel temperature plateau of 500 ± 50°F rated. d. At the end of the first heatup/cooldown cycle (near ambient conditions).

TABLE 14.2-304 (Cont)

2. Record remote sensor displacement readings for: reactor water cleanup, high pressure core spray, low pressure core spray, residual heat removal, ~~control rod drive,~~ reactor core isolation cooling, main steam, ~~main steam safety relief,~~ and feedwater systems.	a. Prior to initial heatup at ambient conditions. b. At the reactor vessel temperature plateau of 300 ± 50°F. c. At the reactor vessel temperature plateau of 500 ± 50°F rated. d. At the end of the first heatup/cooldown cycle (near ambient conditions).
3. Record scriber displacement readings for: reactor water cleanup, reactor core isolation cooling, feedwater, main steam, and condensate systems. a. At the end of the first heatup/cooldown cycle (near ambient conditions) following near rated conditions.	~~a. Prior to initial heatup at ambient~~ conditions. ~~b. At the reactor vessel temperature plateau of 300 ± 50°F.~~ ~~c. At the reactor vessel temperature plateau of 500 ± 50°F rated.~~ ~~d. At the end of first cycle heatup (near ambient conditions).~~
4. Record remote sensor displacements for feedwater system.	a. Upon feedwater system obtaining within ~~±20°F~~ ±35°F of its rated temperature. ~~during TC-6~~

Add items 5-11 from attached sheets

Acceptance Criteria

Level 1:

1. There are no obstructions which will interfere with the thermal expansion of the above piping systems, including snubbers and spring supports attached to the piping systems.

Action	Test Conditions
5. Record selected snubber and spring support deflections for main steam and rx. water cleanup systems.	a. Prior to initial heatup at ambient conditions b. At the reactor vessel temperature plateau of 300 ± 50 °F. c. At the reactor vessel temperature plateau of 500 ± 50 °F. d. At the end of the first heatup/cooldown cycle (near ambient conditions)
6. Record selected snubber and spring support deflections for feedwater, rx. core isolation cooling, and residual heat removal (steam condensing and shutdown cooling modes) systems.	a. Prior to initial system heatup at ambient conditions b. Upon achieving near rated conditions (where accessible). c. At the end of the first heatup/cooldown cycle (near ambient conditions) following near rated conditions.
7. Visual inspection for the condensate system.	a. Prior to initial system heatup at ambient conditions b. At near rated conditions c. At the end of the first system heatup/cooldown cycle (near ambient conditions) following near rated conditions.
8. Record remote sensor displacements for the main steam relief piping	a. Prior to initial system heatup at ambient conditions b. During SRV actuation at TC-1. c. Following SRV actuation at return to near ambient conditions

9. Record remote sensor displacements for residual heat removal system (shutdown cooling mode).	a. Prior to initial system heatup at ambient conditions. b. During shutdown cooling mode operation. c. At the end of the first system heatup / cooldown cycle (near ambient conditions) following operation in the shutdown cooling mode.
10. Record displacements of piping at selected pipe whip restraint locations for feedwater, main steam, rx. core isolation cooling, residual heat removal and rx. water cleanup systems using local measuring devices	a. Prior to initial system heatup at ambient conditions b. At the return to near ambient following achieving near rated conditions for the system
11. Record displacements and perform visual examinations of selected snubbers and spring supports (as defined by the Preservice Inspection Plan).	a. Prior to initial system heatup at ambient conditions. b. At the reactor vessel temperature plateau of 300±50 °F (where applicable to the system or portion thereof involved). c. At the reactor vessel temperature plateau of 500±50°F (where applicable to the system or portion thereof involved). d. At near rated conditions for systems outside the applicability of b. and c. above. e. At the return to near ambient conditions following achieving the applicable near rated conditions (conditions are specified in the test procedure) for the system

TABLE 14.2-304 (Cont)

selected pipe support

2. The displacements at the established transducer, and scriber locations shall not exceed the allowable values provided in the BOP thermal expansion procedure. The allowable values of displacement shall be based on not exceeding ASME Section III code stress allowables. 22

3. Snubbers and spring supports examined shall be found operable as defined by criteria specified in the test procedure.

Level 2:

1. The displacements at established transducer, and scriber locations shall not exceed the expected values as provided in the BOP system expansion procedure.

selected pipe supports, selected whip restraints

2. Selected snubber and spring support examinations shall be evaluated to acceptance criteria defined in the Preservice Inspection Plan for ASME Section IX VT-4 Examinations.

TABLE 14.2-307

DRYWELL HIGH ENERGY PENETRATIONS

Startup Test (SUT-81)

Test Objective

The purpose of this test is to demonstrate the capability of the drywell high energy penetrations to maintain the surrounding concrete below design temperature limits.

Prerequisites

The SORC has reviewed and approved the test procedure and the initiating of testing. Instrumentation has been checked and calibrated as appropriate.

Test Procedure

The following drywell penetration system test will be performed at various power levels during plant startup while at steady-state conditions: 27

1. Temperature - Monitor the thermal rise of the process piping, flued head, and the liner insert junction. ~~Monitor the wall insert annular space if practical.~~

2. The data will then be compared to values predicted for normal operation or for design conditions as required to verify compliance with the acceptance criteria.

Acceptance Criteria

Level 1:

1. The temperatures measured at the junction of the penetrations outer collars and the wall insert sleeves shall not exceed the values predicted to cause surrounding concrete temperatures to exceed 200°F.

Level 2:

1. The temperatures measured at the junction of the penetrations outer collars and the wall insert sleeves shall not exceed their predicted values for normal operating conditions.

TABLE 14.2-240

LOSS OF TURBINE GENERATOR AND OFFSITE POWER

Startup Test (SUT-31)

Test Objective

To determine the electrical equipment and reactor transient performance during the loss of auxiliary power.

Prerequisites

The appropriate preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

The loss of auxiliary power test is performed at 20 to 30 percent of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel generator load are checked. Appropriate reactor parameters are recorded during the resultant transient. The loss of power will be maintained long enough for plant conditions to stabilize (≥30 min). Systems which do not affect vessel level and pressure may be manually started and operated, as necessary.

The following test is performed:

Action	Test Conditions
After transferring auxiliary loads to the unit auxiliary transformer and starting main turbine dc oil pump, use the trip relay to trip the main generator. (SUT-33, Action Item 1, can be done in conjunction with this test.)	a. At TC-2. b. Recirculation system in POS mode. All other systems in NORM mode. 22

TABLE 14.2-240 (Cont)

Acceptance Criteria

Level 1:

1. All safety systems such as the RPS, diesel generators, and HPCS must function properly without manual assistance, and HPCS and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of the LPCS, LPCI, ADS, and MSIV closure. Diesel generators shall start automatically.

22

Level 2:

1. Proper instrument display to the reactor operator shall be demonstrated, including power monitors, pressure, water level, control rod position, suppression pool temperature, and reactor cooling system status. Displays shall not be dependent on specially installed instrumentation.

22 2. If safety/relief valves open, the temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10°F of the temperature recorded before the valve was opened.

TABLE 14.2-243

REACTOR WATER CLEANUP SYSTEM

Startup Test (SUT-70)

Test Objective

To demonstrate specific aspects of the mechanical ability of the RWCU. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating.)

Prerequisites

The preoperational tests have been completed, and the SORC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Procedure

With the reactor at rated temperature and pressure, process variables are recorded during steady-state operation in three modes as defined by the system process diagram: hot standby, normal, and blowdown. A comparison of the bottom head flow indicator and the RWCU inlet flow indicator is made during these modes. The RWCU system sample station is tested at hot process conditions as part of SUT 1. 22

The following test is performed:

Action	Test Conditions
Take heat balance and pressure data.	a. Reactor at rated temperature and pressure during heatup. b. Cleanup system operate in hot standby, normal, and blowdown modes.

Acceptance Criteria

Level 1:

Not applicable.

TABLE 14.2-243 (Cont)

Level 2:

1. The temperature at the tube side of the nonregenerative heat exchangers does not exceed 130°F in the blowdown mode or 120°F in the normal mode.

2. The pump available NPSH at least 13 ft during the hot standby mode is as defined in the process diagrams.

3. The cooling water supplied to the nonregenerative heat exchangers shall be less than 6 percent above the flow corresponding to the heat exchanger capacity (as determined from the process diagram) and the existing temperature differential across the heat exchangers. The outlet temperature shall not exceed 180°F.

22 4. Recalibrate bottom head flow indicator against RWCU flow indicator if the deviation is greater than 25 gpm.

22 5. Pump vibration shall be less than or equal to 2 mils peak-to-peak (in any direction) as measured on the bearing housing, and 2 mils peak-to-peak shaft vibration as measured on the coupling end.