Washington Public Power Supply System
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Docket No. 50-397

January 18, 1985

Mr. John B. Martin, Administrator
Region V Office of Inspection and Enforcement
US Nuclear Regulatory Commission
1450 Maria Lane
Walnut Creek, California 94596

Subject: WASHINGTON NUCLEAR PLANT - UNIT 2
INTERIM STARTUP REPORT

Reference: 1) Plant Technical Specification 6.9.1.1

Reference 1) requires a Startup Report of Plant startup and power ascension testing to be submitted nine (9) months following initial reactor criticality. The first criticality of WNP-2 occurred on January 19, 1984 and a report was submitted on October 18, 1984 which addressed testing through Test Condition No. 1. Subsequent reports are required to be submitted every three months until all testing leading to commencement of commercial operation has been reported.

The purpose of this correspondence is to provide you with the test reports for those tests which FSAR Table 14.2-4 specified to be performed during Test Conditions No. 2 and 3. WNP-2 has completed the testing specified through Test Condition No. 6 and has met all Level 1 acceptance criteria. The results of tests performed subsequent to Test Condition No. 3 are undergoing review and will be the subject of a future final report. This report is being submitted as an interim report and the final report is expected to provide complete information concerning the both interim report's content.

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WASHINGTON NUCLEAR PLANT - UNIT 2
STARTUP REPORT

Attachment A contains the FSAR Chapter 14 test descriptions and test abstracts for Test Condition Nos. 2 and 3. These results have undergone Plant Operations Committee review and our report is based on that review. Please note that the acceptance criteria listed are the criteria for only Test Condition Nos. 2 and 3.

If there are any questions regarding this submittal, please do not hesitate to contact me.

J. D. Martin (M/D 927M)
WNP-2 Plant Manager

JDM:RK:mm

Enclosure:
Attachment A (2 copies)

cc: Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
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Attn: Document Control Desk
Attachment A (36 copies)

TEST NUMBER 1

CHEMICAL AND RADIOCHEMICAL

PURPOSE

The principal objectives of this test are a) to secure information on the chemistry and radiochemistry of the reactor coolant, and b) to determine that the sampling equipment, procedures and analytic techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the reactor system meet specifications and process requirements.

Specific TC-2 & 3 objectives of the test program include evaluation of fuel performance, evaluations of demineralizer operations by direct and indirect methods, measurements of filter performance, confirmation of condenser integrity, demonstration of proper steam separator-dryer operation (TC-2 only), measurement and calibration of the Off-Gas System, and calibration of certain process instrumentation. Data for these purposes is secured from a variety of sources: Plant Operating Records, regular routine coolant analysis, radiochemical measurements of specific nuclides, and specific chemical tests.

CRITERIA

A. LEVEL 1 (TC-2 and TC-3)

Chemical factors defined in the Technical Specifications and Fuel Warranty must be maintained within the limits specified.

The activity of gaseous liquid effluents must conform to license limitations.

Water quality must be known at all times and should remain within the guidelines of the Water Quality Specifications.

B. LEVEL 2 (TC-2 and TC-3)

None

RESULTS

A. TEST CONDITION: TC-2

Testing was performed for stored water quality; reactor water chemistry and radiochemistry, including fuel performance; filter-demineralizer performance; off-gas system performance, including fuel performance; and feed-condensate system performance including condenser integrity. Although reactor water conductivity was sometimes above 1.0 umho/cm, all acceptance criteria were met. The Technical Specification limit for days above 1.0 umho/cm was not exceeded.

A reactor water no cleanup test was conducted at approximately 50% power. All acceptance criteria were met up until the time the test had to be terminated prematurely due to failure of RWCU pump 'A'. The testing scheduled in TC-2 is intended as a dryrun for the final test required at TC-6.

As a result of data taken during testing, design improvements were made to RWCU F/D instrumentation. Various repairs and procedural improvements were made for both the RWCU and the Condensate Filter-Demineralizers. Action was also taken to reduce condenser air in-leakage.

B. TEST CONDITION: TC-3

During TC-3 testing was performed for stored water quality; reactor water chemistry and radiochemistry, including fuel performance; filter-demineralizer performance; off-gas system performance, including fuel performance; and feed-condensate system performance, including condenser integrity. Although reactor water conductivity was sometimes above 1.0 umho/cm all acceptance criteria were met. The Technical Specification limit for days above 1.0 umho/cm were not exceeded.

As a result of data taken during testing, a design change reducing the septum area on RWCU F/D 'A' was implemented. Additional procedural changes were incorporated into the RWCU and condensate filter-demineralizer procedures. Progress was made on the condenser air inleakage problem by repairs to the condensate pump seals, FW heater relief valve tail pipe flanges, and other smaller sources. Several leaking condenser tubes were plugged.

TEST NUMBER 2

RADIATION MEASUREMENTS

PURPOSE

The purposes of this test are a) to determine the background radiation levels in the plant environs prior to operation for base data on activity build-up, and b) to monitor radiation at selected power levels to assure the protection of personnel during plant operation.

CRITERIA

A. LEVEL 1 (TC-2 and TC-3)

The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines for the Standards for Protection Against Radiation outlined in 10CFR20, NRC General Design Criteria.

B. LEVEL 2 (TC-2 and TC-3)

None

RESULTS

A. TEST CONDITION: TC-2 and TC-3

A "Complete Standard Survey" of background radiation levels in the plant environs was performed at 28.6% rated reactor power at TC-2, and another taken at 66% at TC-3. During TC-3 testing it was found that 3 drywell penetrations (X-43B, 94 and 95 at 501' elevation, reactor building) were not sealed against streaming. Although the radiation levels measured at these penetrations were relatively low, administrative controls were established to keep personnel away from them. The penetrations will be sealed at some future date. All other data taken met acceptance criteria.

TEST NUMBER 5

CONTROL ROD DRIVE SYSTEM

PURPOSE

The purpose of the Control Rod Drive (CRD) System test during test conditions 2 and 3 is a) to demonstrate that the CRD system operates properly over the full range of primary coolant temperatures and pressures at normal power operation, and b) to determine the initial operating characteristics of the entire CRD system.

CRITERIA

LEVEL 1 (TC-2 and TC-3)

The scram insertion time of each control rod from full out to position 05, based on deenergization of the scram pilot valve solenoids as time zero, shall not exceed 7.0 seconds.

RESULTS

A. TEST CONDITION: TC-2 and TC-3

Four selected CRDs which had the slowest scram times or unusual operating characteristics during the scram testing performed during Test Condition "Open Vessel", were scram tested in conjunction with PPM 8.2.31 "Loss of Turbine Generator and Offsite Power" and with PPM 8.2.27 "Turbine Trip and Generator Load Rejection". Scram delay times for the four selected CRDs were determined from the scram data and data recorded during pre-operational testing. Scram delay time is defined as the time from a parameter reaching its scram setpoint value until control rods begin their insertion into the core. This may be broken up into the time periods from the parameter reaching its scram setpoint until the scram solenoid valves are de-energized, RPS response time (0.077 seconds for Level Indicating Switches 24A to D), and the time from the scram solenoid valves being de-energized until rod insertion begins (time to notch 47).

The scram times for control rod drives 06-27, 26-55, 30-35, and 34-47 are presented in Tables 1 and 2. All four CRD scram insertion times were less than 7.0 seconds and hence met the Level 1 acceptance criteria.

TABLE 1

Initial Data

This test is being performed in conjunction with the following transient:

PPM 8.2.27 (Test Condition No. 2)

Reactor Pressure 920 psig

CRD Drive Water Pressure Rx + 250 psig

SELECTED CRD LOCATION AND INITIAL POSITION	CNTRL ROD SEO & RSCS GROUP	ACCUMU-LATOR PRESSURE (PSIG)	CRD SCRAM TIME (SEC) TO NOTCH POSITION				SCRAM DELAY TIME
			45	39	25	05	
1) 26-55/48	A	1120	0.302	0.618	1.350	2.522	0.190*
2) 06-27/48	A	1105	0.286	0.638	1.434	2.642	0.178*
3) 34-47/48	A	1110	0.278	0.574	1.242	2.334	0.182*
4) 30-35/48	A	1105	0.294	0.618	1.362	2.510	0.186*
						Ave =	0.182
		SCRAM DELAY TIMES (Including RPS Delay Time)					
26-55							0.267
06-27							0.255
34-47							0.259
30-35							0.263

* Does not include RPS response time of 0.077 seconds

TABLE 2

Initial Data

This test is being performed in conjunction with the following transient:

PPM 8.2.31 (Test Condition No. 3)

Reactor Pressure 963 psig

CRD Drive Water Pressure Rx + 260 psig

SELECTED CRD LOCATION AND INITIAL POSITION	CNTRL ROD SEO & RSCS GROUP	ACCUMU-LATOR PRESSURE (PSIG)	CRD SCRAM TIME (SEC) TO NOTCH POSITION				SCRAM* DELAY TIME
			45	39	25	05	
1) 26-55/48	A	1080	0.302	0.620	1.336	2.470	0.190
2) 06-27/48	A	1095	0.303	0.617	1.399	2.533	0.180
3) 34-47/48	A	1095	0.302	0.586	1.264	2.340	0.189
4) 30-35/48	A	1095	0.297	0.593	1.287	2.341	0.186
						Ave. =	0.186
		SCRAM DELAY TIMES (Including RPS Delay Time)					
26-55							0.267
06-27							0.257
34-47							0.266
30-35							0.263

* Does not include RPS response time of 0.077 seconds

TEST NUMBER 6

SRM PERFORMANCE AND CONTROL ROD SEQUENCE

PURPOSE

The purpose of this test is to demonstrate that the rod withdrawal sequence provides adequate control to increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2

None

RESULTS

A. TEST CONDITION: TC-2

During power ascension to Test Condition 2 (approximately 45% power) control rod sequence A performance and core response was evaluated to be satisfactory.

This data was recorded during power ascension from the Test Condition 1 envelope to Test Condition 2 power levels. Control rods continued to be withdrawn in an A2 control rod sequence. Plant parameters monitored during this test (refer to Table 1) demonstrated that the control rod sequence provides a safe and efficient power ascension between Test Condition 1 and 2 conditions.

TABLE 1

Control Rod Sequence A-2

SEQUENCE STEP COMPLETED	ELECTRICAL POWER MWe	STEAM FLOW Mlb/hr	FW FLOW Mlb/hr	CONTROL VALVE POS % OPEN	APRM READING %					
					A	B	C	D	E	F
23-1	115	2.75	2.30	4.6%	17.5	20	20	21	20	20
23-4	119	2.83	2.45	4.7%	18	20	21	22	21	23
24-6	143	3.10	2.64	5.3%	19.5	24.5	24.5	24	24	24
24-12	148	3.18	2.78	5.4%	19.8	25	24.6	25	24.7	24.6
25-8	212	3.68	3.27	7%	22.6	30.3	29.2	30	30	28
32-12	390	5.5	5.00	12%	47*	47*	46*	47*	47*	48*

* GAF = .94, Actual Power = Reading x .94

TEST NUMBER 10

IRM PERFORMANCE

PURPOSE

The purpose of this test is to adjust the Intermediate Range Monitor System to obtain an optimum overlap with the APRM system.

CRITERIA

A. LEVEL 1

Each APRM must be on scale before the IRM's exceed their rod block setpoint.

B. LEVEL 2

Each IRM channel must be adjusted so that one decade overlap with the APRMs is assured.

RESULTS

TEST CONDITION: TC-2

During Test Condition 2 the LPRM system was recalibrated, thus reverification of the IRM-APRM overlap was required. When the plant was restarted following a scheduled maintenance outage, IRM and APRM readings were recorded periodically as the neutron flux and reactor power increased. The one decade overlap of the Level 2 criteria was met (refer to Table 1).

With the plant heat balance showing 39.35% rated reactor power, IRM's were inserted individually and their reading on range 10 was taken. The IRM readings ranged from 93 on Channel B to 107 on Channel F (refer to Table 2).

TEST NUMBER 10 (Continued)

TABLE 1 IRM/APRM OVERLAP VERIFICATION

IRM Detector	A	B	C	D	E	F	G	H
Reading	40	17	53	60	48	55	40	50
Range	8	7	8	8	8	8	8	8

APRM Channel	A	B	C	D	E	F
Reading %	2	1	2.5	0.8	1.8	1.2

TABLE 2 IRM/THERMAL POWER CORRELATION

IRM Detector	A	B	C	D	E	F	G	H
Reading on Range 10	104	93	100	102	106	107	100	100
Rx Power	39.35%							

TEST NUMBER 11

LPRM CALIBRATION

PURPOSE

The purpose of this test is to calibrate the Local Power Range Monitoring System.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2 (TC-3)

Each LPRM reading will be within 10% of its calculated value.

RESULTS

The LPRM calibration at Test Condition 3 was performed using the process computer program OD-1 "Whole Core LPRM Calibration and Base Distribution". First OD-1 and the P-1 were run to determine the pre-calibration LPRM GAF's. Next the LPRM detector current necessary to produce a 100% reading for that LPRM was determined. This value was then divided by the respective LPRM GAF to produce the calculated current. The final LPRM readings were all within 10% of their calculated values. LPRM 08-41B was bypassed during the calibration because it had failed. LPRM 08-33D was also bypassed during the calibration because it was diagnosed as a drifting LPRM. The LPRM system design allows for numerous LPRM detectors to be bypassed without introducing problems into the process computer power distribution calculations.

TEST NUMBER 12

APRM CALIBRATION

PURPOSE

The purpose of this test is to calibrate the Average Power Range Monitor System.

CRITERIA

A. LEVEL 1(TC-2 and TC-3)

The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

Technical Specification limits on APRM scram and rod block shall not be exceeded.

B. LEVEL 2 (TC-2 and TC-3)

If the above Level 1 criteria are satisfied then the APRM channels will be considered to be reading accurately if they agree with the heat balance or the minimum value required based on peaking factor MLHGR and fraction of rated power to within ± 7% of rated power.

RESULTS

A. TEST CONDITION: TC-2

During Test Condition 2, a reactor heat balance calculation was obtained via OD-3 option-2 (Process Computer Core Thermal Power and APRM calibration). The CMFLPD (Core Maximum Fraction of Limiting Power Density) was obtained via the offline (BUCLE) program. All APRM's were adjusted to read within (+7, -0%) of rated thermal power. The APRM upscale neutron trip for TC-2 was set at 75% of rated. All Level 1 and 2 criteria were satisfied.

B. TEST CONDITION: TC-3

During Test Condition 3 a reactor heat balance calculation was obtained via OD-3 option-2 (Process Computer Core Thermal Power and APRM Calibration). As a result of the periodic surveillance tests performed on the APRM's per Technical Specifications, they were found to be in acceptable agreement with calculated core thermal power and were not adjusted. All Level 1 and 2 criteria were satisfied.

TEST NUMBER 13

PROCESS COMPUTER

PURPOSE

The purpose of this test is to verify the performance of the process computer under plant operating conditions.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2

Program OD-1, P1, and OD-6 will be considered operational when:

1. The MCPR calculated by BUCLE and the process computer either:

   a. Are in the same fuel assembly and do not differ in value by more than 2%, or

   b. For the case in which the MCPR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly, the MCPR and CPR calculated by the two methods shall agree within 2%.

2. The maximum LHGR calculated by BUCLE and the process computer either:

   a. Are in the same fuel assembly and do not differ in value by more than 2%, or

   b. For the case in which the maximum LHGR calculated by the process computer is in a different assembly than that calculated by BUCLE, for each assembly, the maximum LHGR and LHGR calculated by the two methods agree within 2%.

3. The MAPLHGR calculated by BUCLE and the process computer either:

   a. Are in the same fuel assembly and do not differ in value by more than 2%, or

   b. For the case in which the MAPLHGR calculated by the process computer is in different assembly than that calculated by BUCLE, for each assembly, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2%.

TEST NUMBER 13 (Continued)

4. The LPRM gain adjustment factors calculated by BUCLE and the process computer agree to within 2%.

5. The remaining programs will be considered operational upon successful completion of the static and dynamic testing.

RESULTS

TEST CONDITION: TC-2

The Dynamic System Test Case was conducted during Test Condition 2 between approximately 21% and 42% power. To begin the test, plant sensor operability was checked and then the computer was initiated via OD-15. All programs enabled by OD-15 were verified and as a result P4, OD-3, 7, 8, 15, 18, 19 and 20 were determined to be operational. LPRM calibration, power distribution and core limits calculations were verified next, and OD-1, 10, and 16 were found to be operational. The daily, monthly and security logs as well as the LPRM sensitivity programs were examined to ensure operability of P2, P3, and OD-13. Fuel bundle locations and exposures were obtained for site verification and for transmittal to General Electric. OD-15 Computer Outage Recovery Monitor (CORM) and OD-15 security log were verified as two means of restoring the process computer following an outage at steady-state power. Finally, the LPRM digital filtering and drift diagnostic program was verified. Programs P5, OD-12, 14 and 17 were determined to be operational. Programs P-1 and OD-6 were considered operational after a comparison of the thermal limits calculated by these programs with those calculated by the Backup Core Limits Evaluation (BUCLE) program showed excellent agreement.

TEST CONDITION: TC-3

Programs OD-2, 4, 5, 9 and 11 were not tested during Test Condition 3. Instead, these programs are to be verified during Test Condition 6. The testing was successful and the programs considered operational.

TEST NUMBER 14

REACTOR CORE ISOLATION COOLING SYSTEM

PURPOSE

The purpose of this test is to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) system over its expected operating pressure range.

CRITERIA

A. LEVEL 1 (TC-2)

The average pump discharge flow must be equal to or greater than 600 gpm after thirty seconds have elapsed from automatic initiation at any reactor pressure between 150 psig and rated.

The RCIC turbine must not trip off or isolate during auto or manual start tests.

B. LEVEL 2 (TC-2)

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The differential pressure switch for the RCIC steam supply line high flow isolation trip shall be adjusted to actuate at the value specified in Plant Technical Specification (About 300%).

The speed and flow control loops shall be adjusted so that the decay ratio of any RCIC system related variable is not greater than 0.25.

In order to provide an overspeed trip avoidance margin, the transient start first and subsequent speed peaks shall not exceed 5% above the rated RCIC turbine speed.

RESULTS

A. TEST CONDITION: TC-2

Test Condition 2 testing of the RCIC system involved two cold quick starts with the reactor at 150 psig and 41% rated power respectively in the CST to CST test mode. The tests demonstrated reliable automatic initiation, reliable continuous operation at rated flow conditions, and automatic transfer of pump suction from the condensate storage tank to the suppression pool. The RCIC turbine gland seal system demonstrated the capability of preventing steam leakage to the atmosphere. The RCIC rated pressure steam flow was evaluated by measuring the bypass valve position before and during RCIC system operation. This data was used to verify the RCIC steam supply line high flow isolation trip setting. All acceptance criteria for TC-2 were met.

TEST NUMBER 16A

SELECTED PROCESS TEMPERATURES

PURPOSE

The purpose of this procedure is to a) verify the setting of the low flow control limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region, b) assure that the measured bottom head drain temperature corresponds to the bottom head coolant temperature during normal operations, and c) identify any reactor operating modes during recirculation pump restarts or one pump operation that cause temperature stratification.

CRITERIA

A. LEVEL 1 (TC-3)

1. The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F.

2. The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F of the steam dome temperature. If two pumps are idle, the loop suction temperature must be within 50°F of the steam dome temperature before pump startup.

B. LEVEL 2 (TC-3)

During two-pump operation at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F of the recirculation loop temperatures.

RESULTS

A. TEST CONDITION: TC-2

With the reactor in a steady state condition near rated temperature and pressure at about 42% rated reactor power, data was taken to verify the absence of temperature stratification in the bottom head region. The recirculation pumps were on 60 Hz with flow control valve 'A' at 7% open and 'B' at 20% open position. Minimal temperature stratification was observed during TC-2.

TEST NUMBER 16A (Continued)

B. TEST CONDITION: TC-3

During Test Condition 3 selected process temperatures were monitored prior to and after individual recirculation pump trips and a two recirculation pump trip (to LFMG).

The reactor coolant temperature difference between the steam dome and bottom head drain were less than 145°F during all tests, satisfying the Level 1 acceptance criteria. The maximum temperature difference obtained during this test was 47°F after the first trip of recirculation pump 'B'.

After the trip of recirculation pump 'B', the maximum difference between the 'A' loop (active) suction temperature and the idle 'B' loop suction temperature was 5°F. After the two pump trip, the maximum temperature difference between the recirculation loop suction temperature of each idle loop and the steam dome saturation temperature was 21°F for both the 'A' and 'B' loop.

During two pump operation at rated core flow, the bottom head coolant temperature, as measured by the bottom drain line thermocouple, was 14°F less than the loop 'A' suction temperature and 16°F less than the loop 'B' suction temperature. Hence, all acceptance criteria were met.

TEST NUMBER 16B

WATER LEVEL REFERENCE LEG TEMPERATURE MEASUREMENT

PURPOSE

The purpose of this test is to measure the reference leg temperature and recalibrate the affected level instruments if the measured temperature is different than the value assumed during the initial calibration.

CRITERIA

A. LEVEL 1 (TC-2 and TC-3)

None

B. LEVEL 2 (TC-2 and TC-3)

The indicator readings on the narrow range level system should agree within ± 1.5 inches of the average readings or the reading calculated from the correct reference leg temperatures.

The wide and upset range level system indicators should agree within ± 6 inches of the average readings or the readings calculated from the correct reference leg temperatures.

RESULTS

A. TEST CONDITION: TC-2

Temperatures in the area of the reference legs of the water level instruments as well as water level instrument readings were taken at steady state rated conditions during Test Condition 2. All temperatures taken were within the calibration tolerance. However, eight instruments consisting of two narrow range and six wide range, failed to meet the Level 2 acceptance criteria. All narrow range instruments were within 2 inches of the criteria range while all wide range and upset level instruments were within 4 inches of the criteria range. An investigation was initiated to determine if the reference leg condensing pot elevations of these instruments was the cause. No calibration data adjustments were necessary as a result of the investigation. The problem was determined to be caused by the relatively long time frame involved for data collection, the need for additional attention during the calibration of the LITS indication and the fluctuation in level from hydraulic effects. More attention was given to calibrating the indicator and additional manpower was assigned during data collection.

TEST NUMBER 16B (Continued)

B. TEST CONDITION: TC-3

Temperatures in the area of the reference legs of the water level instrument as well as water level instrument readings were taken at steady state rated conditions during Test Condition 3. All temperatures taken were within the calibration tolerance. Reactor water level readings were taken at steady state conditions. The readings resulted in all level indicators meeting the acceptance criteria.

TESTS NUMBER 17 & 33

SYSTEM EXPANSION & VIBRATION

PURPOSE

The purpose of test 17 is to a) verify that piping systems and components are unrestrained with respect to thermal expansion, b) verify that suspension components are functioning in the specified manner, c) provide confirmatory data for the calculated stress levels in nozzles and weldments, d) perform an inspection to satisfy ASME Section XI, IWF-220 post heatup (shakedown) inspection requirements, and e) satisfy the inspection requirements for the condensate and feedwater systems per Regulation Guide 1.68.1.

The purpose of test 33 is to verify that the design stress levels due to piping vibration are not exceeded and satisfy the inspection requirements for condensate and feedwater systems per Regulation Guide 1.68.1.

CRITERIA

A. LEVEL 1 (Test 17)

Thermally induced displacement of system components shall be unrestrained, with no evidence of binding or impairment. Spring hangers shall not be bottomed out or have the spring stretched.

Snubbers shall not reach the limits of their travel. The displacements at the established transducer locations used to measure pipe deflections shall not exceed the allowable values. The allowable values of displacement shall be based on not exceeding ASME Section III Code Stress allowables.

B. LEVEL 2 (Test 17)

Spring hangers will be in their operating range (between the hot and cold settings).

Snubber settings must be in operating range.

The displacements at the established transducer locations shall not exceed the expected values.

C. LEVEL 1 (Test 33)

The measured vibration amplitude (peak-to-peak) of the systems monitored shall not exceed the maximum allowable displacements.

D. LEVEL 2 (Test 33)

The measured amplitude (peak-to-peak) of vibration shall not exceed the expected values.

TEST NUMBERS 17 & 33 (Continued)

RESULTS

A. The following tests were performed at TC-2:

1. Steady state instrumented vibration for main steam and feedwater piping inside the drywell at about 25% power.

2. Steady state visual exam of condensate and feedwater piping outside the drywell at about 25% power.

3. Instrumented vibration of main steam and recirculation system piping during the generator load reject test at 25% power.

4. Steady state instrumented vibration for main steam, feedwater, and recirculation system piping inside the drywell at 50% power.

5. Main steam and SRV tailpipe instrumented vibration during SRV capacity testing.

6. Main turbine bypass valve piping visual exam during SRV capacity testing.

All test data taken was well within the acceptance criteria. The visual examinations provided acceptable results.

B. The following tests were performed at TC-3:

1. Steady state instrumented vibration for main steam, feedwater, and recircuation system piping inside the drywell at 75% power.

2. Recirculation piping instrumented vibration during a single recirculation pump trip from 75% power and 100% core flow, and during the subsequent recirculation pump restart.

3. Recirculation piping instrumented vibration during a simulated T/G trip (by opening the RPT breakers) tripping both recirculation pumps to 15 Hz from 75% power and 100% core flow.

4. Main steam and recirculation system instrumented vibration during a main turbine trip from 75% power.

All test data taken was within the acceptance criteria.

TEST NUMBER 18

CORE POWER DISTRIBUTION

PURPOSE

The purpose of this test is to determine the reproducibility of the TIP system readings.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2

The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the uncertainties for all data sets shall be less than 6.0%.

The data acquired for random noise uncertainty does not have specific acceptance criteria value and is used only to aid in the analysis of the TIP uncertainty.

RESULTS

During TC-3, Traversing In-Core Probe (TIP) system readings reproducibility and Core Power Distribution were evaluated with the aid of the GE Mark III program "BILLEXE". The total uncertainty in the TIP system was found to be 3.20%, well below the maximum Level 2 criteria requirement of 6.0%. Core power distribution was examined and was determined to be symmetric.

TEST NUMBER 19

CORE PERFORMANCE

PURPOSE

The purpose of this test is to a) evaluate the core thermal power and b) evaluate the following core performance parameters are within limits: 1) maximum linear heat generation rate, 2) minimum critical power ratio (MCPR), and 3) maximum average planar linear heat generation rate (MAPLHGR).

CRITERIA

A. LEVEL 1

The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady state conditions shall not exceed the limit specified by the Plant Technical Specifications.

The steady state Minimum Critical Power Ratio (MCPR) shall not exceed the minimum limits specified by the Plant Technical Specifications.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits specified by the Plant Technical Specifications.

Steady state reactor power shall be limited to the rated MWT and values on or below the design flow control line. Core flow shall not exceed its rated value.

B. LEVEL 2

None

RESULTS

A. TEST CONDITION: TC-2

Core Thermal Power (CTP) was evaluated by a manual heat balance calculation, and core performance parameters (MLHGR, MCPR, and MAPLHGR) were obtained from the GE off-line computer program BUCLE using the TIPNEWRP & PINEWRP options. The core performance parameters were found to be within the acceptance criteria.

B. TEST CONDITION: TC-3

All core performance parameters, including core flow and CTP, were calculated by the process computer and obtained from edits of programs P1, OD-3, and OD-6. All test criteria were satisfied.

TEST NUMBER 22

PRESSURE REGULATOR

PURPOSE

The purpose of this test is to a) determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of pressure regulators, b) demonstrate the backup capability of the pressure regulators via simulated failure of the controlling pressure regulator and to set the regulating pressure difference between the two regulators at an appropriate value, c) demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine, and d) demonstrate that affected parameters are within acceptable limits during pressure regulator induced transient maneuvers.

CRITERIA

A. LEVEL 1

The transient response to any pressure control system related variable to any test input must not diverge.

B. LEVEL 2

1. Pressure control 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.

2. The turbine inlet pressure response time from initiation of pressure setpoint change to the turbine inlet pressure peak shall be 10 seconds.

3. Pressure control system deadband, decay, etc., shall be small enough that steady state limit cycles (if any) shall produce steam flow variations no larger than ± 0.5 percent of rated steam flow.

4. For all pressure regulator transients the peak neutron flux and/or peak vessel pressure shall remain below the scram settings by 7.5% and 10 psi respectively (maintain a plot of power versus the peak variable values along the 100% rod line).

TEST NUMBER 22 (Continued)

5. The variation in incremental regulation (ratio of the maximum to the minimum value of the quantity, "incremental change in pressure control signal/incremental change in steam flow", for each flow range) shall meet the following:

% of Steam Flow Obtained With Valves Wide Open	Variation
0 to 90	4:1
90 to 97%	2:1
90 to 99%	5:1

C. LEVEL 3

1. Additional dynamics of the control system outside of the regulator compensation filters, shall be equivalent to a time constant no greater than 0.10 second. This also includes any dead time which may exist.

2. Control by bypass valve motion must respond to pressure inputs with deadband (insensitivity) no greater than ± 0.1 psi.

3. Dynamics of both pressure regulators will be essentially identical.

RESULTS

A. TEST CONDITION: TC-2

Pressure regulator setpoint step changes and auto transfer testing was performed at approximately 27% power with the turbine generator on line for each operating mode: GV, BPV Incipient, and BPV. All acceptance criteria were satisfied with the exception of several cases of decay ratios exceeding the 0.25 Level 2 limit. Since in all cases the oscillations decayed away within 4 cycles, indicating a stable system, performance was evaluated to be adequate to proceed to TC-3.

B. TEST CONDITION: TC-3

Pressure regulator setpoint step changes and auto transfer testing was performed at approximately 63.5% power for the GV operating mode. All acceptance criteria, including the 0.25 Level 2 decay ratio, were satisfied.

During the ascension to TC-3, pressure regulator linearity readings were recorded approximately every 2% increase in reactor thermal power. Several of the readings taken were considered suspect, and it is expected that an accurate set of readings will be taken during the ascension to TC-6. Nevertheless, the maximum variation in incremental regulation was calculated to be 3.96:1, which meets the acceptance criterion.

TEST NUMBERS 23A & D

FW SYSTEM WATER LEVEL SETPOINT & MANUAL FLOW CHANGES AND MAXIMUM FEEDWATER RUNOUT CAPABILITY

PURPOSE

The purpose of test 23A is to verify that the feedwater system has been adjusted to provide acceptable reactor water level control.

The urpose of this portion of test 23D is to calibrate the feedwater cont ols.

CRITERIA

A. LEVEL 1

The transient response of any level control system related variable to any test must not diverge.

B. LEVEL 2

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

2. The open loop dynamic flow response of each feedwater actuator (turbine or valve) to small ($<$10%) step disturbances shall be:

a.	Maximum time to 10% of step disturbance	$\leq$ 1.1 sec
b.	Maximum time from 10% to 90% of a step disturbance	$\leq$ 1.9 sec
c.	Peak overshoot (% of step disturbance)	$\leq$ 15%
d.	Settling time, 100%, $\pm$ 5%	$\leq$ 14 sec

3. The average rate or response of the feedwater actuator to large (20% of pump flow) step disturbances shall be between 10% and 25% rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10% and 90% response points, TC-3 only.

C. LEVEL 3

1. The dynamic response of each individual level or flow shall be as fast as possible. Band width must be at least 4.0 radians/second (faster than 0.25 second equivalent time constant), except for the steam flow sensors which must have band width of at least 1.0 radian/second (faster than 1.0 second equivalent time constant), TC-2 only.

TEST NUMBERS 23A & D (Continued)

2. Vessel level, feedwater flow, and steam flow sensors must be installed with sufficiently short lines and proper damping adjustment so that no resonances exist, TC-2 only.

3. Initial settings of the function generators should give a straight line. The function generators must be adjusted so that the change in slope (actual fluid flow change divided by demand change for small disturbances) shall not exceed a factor of 2 to 1 (maximum slope versus minimum slope) over the entire 20% to 100% feed flow range. Also the function generators should be used to minimize the differences between feedwater actuators (pumps and/or valves), TC-2 only.

4. All auxiliary controls which have direct impact on reactor level and feedwater control (e.g., feed pump minimum recirculation flow valve control) should be functional, responsive, and stable. The minimum low valve control should be fast enough to avoid pump trips and yet slower than the feedwater startup valve to avoid possible reactor flux scram due to a cold water slug.

RESULTS

A. TEST CONDITION: TC-2

Instrument response for the Level 3 acceptance criteria was measured during the preoperational test program with satisfactory results. Startup level controller response adjustment was done during TC-Heatup and TC-1.

TC-2 testing consisted of 3" and 6" level step changes in both directions with the master controller in both the single and three element modes, and of manual feedwater flow changes. All acceptance criteria were satisfied with the following exceptions:

1. Open loop testing for dynamic flow response (<10%) flow disturbance on startup level control valve FCV-10 was not done and was rescheduled for another test condition.

2. Undershoot was observed during manual flow steps for RFW turbine 'A' due to a faulty governor valve servo motor. The servo was replaced during the outage between TC-2 and TC-3 and retested in TC-3.

3. The manual flow steps were performed with a 16% flow step change instead of 10% as required by the test procedure. The larger steps were adequate to evaluate the controller response.

TEST NUMBERS 23A & D (Continued)

As a result of data taken during TC-2 testing the following actions were taken:

1. Procedural improvements were implemented for transfer of level control from FCV-10 to TDRFP speed control.

2. Due to a flow capacity mismatch between the startup level controller and the master controller, the minimum speed of the feedwater pump turbines was reduced to 2500 rpm. This allowed successful transfer from the startup level control valve to the feedwater turbine speed control.

B. TEST CONDITION: TC-3

Since TC-3 is the power/flow condition that is the least stable, the final proportional gain setting was established by performing the calibration portion of 23D in conjunction with the applicable steps in 23A during ascension to TC-3.

TC-3 testing consisted of 3" and 6" level step changes in both directions with the master controller in single element and again in three element, with recirculation in both position and flux modes; and of manual flow small (< 10%) and large (> 20%) step changes. All acceptance criteria were met with the exception of some peak overshoot (> 15%) for open loop dynamic flow response of the 'A' feedwater turbine to small flow step changes. This exception was considered acceptable for current operation because the reactor vessel level was adequately controlled in both the single and three element modes.

TEST NUMBER 25

MAIN STEAM ISOLATION VALVES

PURPOSE

The purpose of this test is to a) functionally check the main steam line isolation valves (MSIVs) for proper operation at selected power levels, b) determine isolation valves closure times at rated conditions, and c) determine the maximum power at which a single valve closure can be made without scram.

CRITERIA

A. LEVEL 1 (Individual Valve Closure)

MSIV closure time, exclusive of electrical delay, shall be no faster than 3.0 seconds (average of the fastest valve in each steam line) and no slower than 5.0 seconds (each valve, not averaged). The electrical time delay at 100% open shall be less than or equal to 0.5 seconds and the fastest valve closure time shall be $\geq$ 2.5 seconds.

B. LEVEL 2 (Individual Valve Closure)

During full closure of individual valves peak vessel pressure must be 10 psi (0.7 kg/cm^2) below scram setpoint, peak neutron flux must be 7.5% below scram setpoint, and steam flow in individual lines must be 10% below the isolation trip setting. The peak heat flux must be 5% less than its trip point.

The reactor shall not scram or isolate.

RESULTS

A. TEST CONDITION: TC-2

Individual main steam isolation valve (MSIV) functional tests were performed at rated reactor temperature and pressure during TC-2 to check for proper valve operation. Valve fast closure times were determined during a manual isolation while at rated conditions. All valves were found to meet the acceptance criteria. Refer to Table 2 for the MSIV fast closure times.

TEST NUMBER 25 (Continued)

TABLE 2

MSIV FAST CLOSURE TIMES

VALVE	CLOSURE TIME *
MS-V-22A	3.77 seconds
MS-V-22B	4.01 seconds
MS-V-22C	4.10 seconds
MS-V-22D	3.89 seconds
MS-V-28A	3.41 seconds
MS-V-28B	4.16 seconds
MS-V-28C	3.21 seconds
MS-V-28D	3.34 seconds

Criteria: MSIV closure time, exclusive of electrical delay, shall be no faster than 3.0 seconds (average of the fastest valve in each steam line) and no slower than 5.0 seconds.

*Exclusive of electrical delay.

TEST NUMBER 26

RELIEF VALVES

PURPOSE

The purpose of this test is to a) verify the proper operation of the main steam relief valves, b) verify the discharge piping is not blocked, c) verify their proper seating following operation, d) obtain transient recorder signature information of relief valve response for subsequent comparisons, and e) determine relief valve capacities.

CRITERIA

A. LEVEL 1

There should be positive indication of steam discharge during the manual actuation of each valve.

The sum total of the capacity measurements from all relief valves shall be equal to or greater 15.8×10^6 lb/hr at an inlet pressure of 103% of 1205 psig.

The total flow capacity of the safety relief valves used in the Automatic Depressurization system must be equal to or greater than 4.8×10^6 lb/hr at 1125 psig when the valve having the highest measured capacity is assumed to be out of service.

B. LEVEL 2

Relief valve leakage shall be low enough that the temperature measured by the thermocouples in the discharge side of the valves returns to within 10°F of the temperature recorded before the valve was opened.

The pressure regulator must satisfactorily control the reactor transient and close the control valves or bypass valves by an amount equivalent to the relief valve discharge.

Each relief valve shall have a capacity between 90% and 122.5% of its expected flow rate of 906,200 lbs/hr at 103% of the inlet pressure of 1205 psig.

No more than 25% of the relief valves may have an individual corrected flow rate that is less than their expected flow rate.

The transient recorder signatures for each relief valve must be analyzed for relative system response comparison.

TEST NUMBER 26 (Continued)

RESULTS

A. TEST CONDITION: TC-2 and TC-3

Proper operation of the Safety Relief Valves was verified by demonstrating that relief valve steam was discharged to the suppression pool and that the valves reseated after actuation. This was accomplished by cycling each relief valve individually and recording discharge line (tail pipe) thermocouple readings prior to and after relief valve actuation. In addition, acoustical monitors were used to indicate the discharge of steam to the suppression pool and the reseating of the relief valves.

The transient recorder signature for each relief valve was analyzed satisfactorily.

The pressure regulator satisfactorily controlled the reactor pressure transient during the actuation of relief valves.

Incremental change in bypass valve position was first correlated with the corresponding change in feedwater flow. Since a change in feedwater flow relates directly to a change in steam flow, the capacity of each relief valve was determined by the change in bypass valve position during relief valve actuation. The steam flow through each relief valve met the acceptance criteria. The sum of the relief valve capacity met the acceptance criteria.

The total SRV flow (corrected from 932 psia to 1256.3 psia) was 18,345,000 #/hr. MS-RV-2D failed the L2 criteria for individual valve capacity. The valve is not an ADS/SRV and all the L1 criteria were met. The flow shortfall was evaluated as acceptable. Subsequent inspections were performed on the valve stroke and the sparger for debris. No cause was found for the flow shortfall. The valve has been scheduled to be in the first batch tested per ASME Section XI in an attempt to determine the cause.

The total ADS/SRV flow equalled 5,713,978 #/hr (corrected to 1139.7 from 932 psia).

TEST NUMBER 27

TURBINE TRIP AND GENERATOR LOAD REJECTION

PURPOSE

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

CRITERIA

A. LEVEL 1 (TC-2)

None

B. LEVEL 1 (TC-3)

For turbine and generator trips from power levels greater than 50% rated reactor power, there should be a delay of less than 0.1 second following the beginning of turbine control or stop valve closure before the beginning of bypass valve opening. The bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of their capacity within 0.3 seconds from the beginning of control or stop valve closure motion.

Feedwater system settings must prevent flooding of the steam line following these transients.

The two recirculation pump drives flow coastdown transient during the first six seconds must be equal to or faster than that specified in Power Ascension Test 8.2.30.B.

The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.

The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of the rated value.

The total time delay from start of turbine stop valve motion or control valve motion to the complete suppression of electrical arc between the fully open contacts of the RPT circuit breakers shall be less than 190 milliseconds.

C. LEVEL 2 (TC-2)

For the generator trip within the bypass valves capacity, the reactor shall not SCRAM for initial thermal power values within that bypass valve capacity.

Electrical load transfers occur as designed.

The measured bypass capacity (in percent of rated power) shall be equal or greater than 3,567,000 lb/hr.

TEST NUMBER 27 (Continued)

D. LEVEL 2 (TC-3)

There shall be no MSIV closure during the first three minutes of the transient and operator action shall not be required during that period to avoid the MSIV trip.

The positive change in the vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

Recirculation LFMG sets shall take over after the initial recirculation pump trips and adequate vessel temperature difference shall be maintained.

Feedwater level control shall avoid loss of feedwater due to possible high level (L8) trip during the event.

Low water level (L2) total recirculation pump trips, HPCS and RCIC shall not be initiated.

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. In addition the acoustical monitors should indicate the valve(s) is closed after the transient is complete.

Electrical load transfers occur as designed.

RESULTS

A. TEST CONDITION: TC-2

The TC-2 generator trip was performed at 25% rated thermal power, which is within the bypass valve rated capacity. The Level 2 acceptance criteria that the reactor shall not SCRAM and the electrical load transfers occur were both satisfied. The bypass valve capacity was measured by a special test procedure and determined to meet the Level 2 acceptance criteria.

TEST NUMBER 27 (Continued)

B. TEST CONDITION: TC-3

The TC-3 main turbine trip was performed at 68% rated thermal power by initiating a manual turbine trip. The resulting transient involved turbine stop valve/control valve fast closure, reactor SCRAM, recirculation pump trip and transfer of auxiliary load to the start-up transformers. All the acceptance criteria were met, except the Level 2 acceptance criteria that feedwater level control shall avoid feedwater pump trip on high reactor water level was not tested. A temporary feedwater procedure in place at the time of the main turbine trip required one feedwater pump to be tripped manually following a SCRAM on turbine trip to guarantee the high water level trip of both feedwater pumps did not occur. Excessive feedwater piping movement during previous scrams indicated the transient to be initiated by the restart of the Feedwater System following a scram. The feedwater turbine trip had occurred previously due to the L8 trip as a result of the system rapid response to the initial indicated level shrink from the void collapse and subsequent swell due to the overfill, cold feedwater expansion and reactor depressurization. The actions taken in the procedure were selected to enable an evaluation of the type of level setdown features desired from an existing design being considered for implementation. Although the feedwater pump trip acceptance criteria could not be properly evaluated due to the procedure actions, the consequences of possible undue feedwater movement and piping thermal stress was considered to be of primary importance.

Data reduction to assess the LI criteria requiring 80% of the design bypass valve capacity within .3 seconds from the beginning of control or stop valve movement indicated unexpected valve movement. Extensive "cold" testing during the plant outage following the turbine trip test at 75% power resulted in system modifications that provided acceptable BPV response. Additional "hot" testing following the outage verified acceptable performance prior to exceeding 25% reactor power. BPV response will be reverified during the Turbine Load Reject Test at 100% power.

TEST NUMBER 29

RECIRCULATION FLOW CONTROL

PURPOSE

The purpose of this test is to a) demonstrate the core flow system's control capability over the entire flow control range, including valve position, core flow, and neutron flux modes of operation, and b) determine that all electrical compensators and controllers are set for desired system performance and stability.

CRITERIA

A. VALVE POSITION CONTROL

LEVEL 1

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

LEVEL 2

Recirculation 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.

The maximum rate of change of valve position shall be 10 ± 1%/sec. The overshoot after a small position demand input (1% to 5%) step shall be < 10% of magnitude of input.

Gains shall be set to give as fast a response as possible to achieve a rise time of ≤ 0.45 seconds for large position demand step inputs (0.5% to 5%) and ≤ 0.25 seconds for small position demand step inputs (0.2% to 0.5%). The delay time should be ≤ 0.15 seconds for large position demand step inputs (0.5% to 5%) and ≤ 0.25 for small position demand step inputs (0.2% to 0.5%).

LEVEL 3

Gains shall be adjusted to give the fastest possible response within the 10% overshoot criteria above and without additional valve duty cycle.

Position loop deadband shall be < 0.2% of full valve stroke.

B. FLOW LOOP CONTROL

LEVEL 1

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

TEST NUMBER 29 (Continued)

LEVEL 2

The decay ratio of the flow loop response to any test inputs shall be 0.25.

The flow loops provide equal flows in the two loops during steady state operation. Flow loop gains should be set to correct a flow imbalance in less than 25 seconds.

The delay time for flow demand step ($\leq$5%) shall be 0.4 seconds or less.

The response time for flow demand step ($\leq$5%) shall be 1.1 seconds or less.

The maximum allowable flow overshoot for step demand of $\leq$ 5% of rated shall be 6% of the demand step.

The flow demand step settling time shall be $\leq$ 6 seconds.

LEVEL 3

Incremental gain from function generator for valve position demand input to sensed drive flow shall not vary by more than 2 to 1 over the entire flow range.

Flow loop upper limit shall be properly set.

C. FLUX LOOP

LEVEL 1

The flux loop response to test inputs shall not diverge.

LEVEL 2

Flux overshoot to a flux demand step shall not exceed 2% of rated for a step demand of $\leq$ 20% of rated.

The delay time for flux response to a flux demand step shall be $\leq$ 0.8 seconds.

The response time for flux demand step shall be $\leq$ 2.5 seconds.

The flux settling time shall be $\leq$ 15 seconds, for a flux demand step $\leq$ 20% of rated.

D. SCRAM AVOIDANCE

LEVEL 1

None

TEST NUMBER 29 (Continued)

LEVEL 2

For any one of the above loops' test maneuvers, the trip avoidance margins must be at least the following:

1. For APRM $\geq$ 7.5%

2. For simulated heat flux $\geq$ 5.0%

E. FLUX ESTIMATOR

LEVEL 1

None

LEVEL 2

Switching between estimated and sensed flux should not exceed 5 times/5 minutes at steady state.

During flux step transient there should be no switching to sensed flux or if switching does occur, it should switch back to estimated flux within 20 seconds of the start of the transient.

F. FLOW CONTROL VALVE DUTY CYCLE

LEVEL 1

None

LEVEL 2

The flow control valve duty cycle in any operating mode shall not exceed 0.2% Hz.

RESULTS

A. TEST CONDITION: TC-3

The recirculation flow control system testing was performed during TC-3 along the 75% rod line.

The TC-3 phase of the recirculation flow control system tests verified that acceptable gain settings exist for valve position, core flow, and neutron flux modes of operation over the entire flow control range with the recirculation pumps operating on high speed.

TEST NUMBER 29 (Continued)

All Level 1 acceptance criteria were met. All Level 2 and 3 acceptance criteria were met except the delay and response time criteria for valve position control and flow loop control were slightly exceeded. The amount of the deviation is considered acceptable. The actual delay and response times are listed on Tables 5, 6 and 7.

The L2 scram avoidance margin was not met in the flow control mode. A circuit modification affecting the sample and hold cards is scheduled during the next outage which is expected to eliminate the problem. Testing scheduled in TC-6 is expected to verify acceptable performance.

TEST NUMBER 29 (Continued)

TABLE 5

VALVE POSITION CONTROL DATA 'A' LOOP

INITIAL VALVE POSITION %	STEP SIZE %	DELAY TIME (sec) * 0.15	RESPONSE TIME (sec) * 0.45	% OVER-SHOOT * 10%	COMB. DELAY & RESPONSE * 0.6
15	0.5(D)	0.2	0.35	10	0.55
15	0.5(U)	0.2	0.3	10	0.5
15	1(D)	0.2	0.2	13.9	0.4
15	1(U)	0.2	0.2	11	0.4
15	5(D)	0.2	0.5	4	0.7
15	5(U)	0.2	0.5	3	0.7
25	0.5(D)	0.2	0.45	25	0.65
25	0.5(U)	0.2	0.3	7	0.5
25	1(D)	0.15	0.2	13.3	0.35
25	1(U)	0.15	0.2	13.3	0.35
25	5(D)	0.2	0.55	3	0.75
25	5(U)	0.15	0.6	3	0.75
50	0.5(D)	0.2	0.1	20	0.3
50	0.5(U)	0.2	0.2	20	0.4
50	1(D)	0.2	0.4	16.6	0.6
50	1(U)	0.2	0.2	21.7	0.4
50	5(D)	0.2	0.55	5	0.75
--	5(D)	0.2	0.44	4.5	0.64
50	5(U)	0.2	0.6	4	0.8
--	----	0.2	0.44	4.5	0.64
75	0.5(D)	0.2	0.2	20	0.4
75	0.5(U)	0.3	0.3	13	0.6
75	1(D)	0.2	0.2	10	0.4
75	1(U)	0.2	0.2	16	0.4
75	5(D)	0.2	0.55	3	0.75
75	5(U)	0.2	0.6	3	0.8

*Acceptance Criteria
(D) = down; (U) = up
--- indicates data inclusive

TEST NUMBER 29 (Continued)

TABLE 6

VALVE POSITION CONTROL DATA 'B' LOOP

INITIAL VALVE POSITION %	STEP SIZE %	DELAY TIME (sec) * 0.15	RESPONSE TIME (sec) * 0.45	% OVER-SHOOT * 10%	COMB. DELAY & RESPONSE * 0.6
15	0.5(D)	0.4	0.5	13.3	0.9
15	0.5(U)	0.5	0.4	33.3	0.9
15	1(D)	0.25	0.3	17	0.55
15	1(U)	0.25	0.3	20	0.55
15	5(D)	0.2	0.4	5.0	0.6
15	5(U)	0.2	0.4	5.0	0.6
25	0.5(D)	0.5	0.3	10	0.8
25	0.5(U)	0.4	0.3	10	0.7
25	1(D)	0.2	0.3	10	0.5
25	1(U)	0.3	0.2	17	0.5
25	5(D)	0.2	0.4	4	0.6
25	5(U)	0.2	0.4	4	0.6
50	0.5(D)	0.4	0.3	10	0.7
50	0.5(U)	0.4	0.3	10	0.7
50	1(D)	0.3	0.35	10	0.65
50	1(U)	0.3	0.2	17	0.5
50	5(D)	0.2	0.4	4	0.6
50	5(D)	0.2	0.4	4	0.6
75	0.5(D)	0.2	0.4	10	0.6
75	0.5(U)	0.3	0.3	10	0.6
75	1(D)	0.2	0.3	10	0.5
75	1(U)	0.2	0.3	12.5	0.5
75	5(D)	0.2	0.4	2	0.6
75	5(U)	0.2	0.4	2	0.6

*Acceptance Criteria
(D) = down; (U) = up
--- indicates data inclusive

TEST NUMBER 29 (Continued)

TABLE 7

FLUX MANUAL (FLOW AUTO) DEMAND STEP DATA

RECIRC LOOP	INITIAL VALVE POSITION %	STEP SIZE %	DELAY TIME (Sec) * 0.4	RESPONSE TIME (Sec) * 1.1	% OVER-SHOOT * 0.6	COMB. DELAY & RESPONSE 1.5
A	15	5(D)	0.5	0.9	0	1.4
A	15	5(U)	0.5	1.1	---	1.6
B	15	5(D)	0.8	1.4	0.6	2.2
B	15	5(U)	0.8	2.0	---	2.6

*Acceptance Criteria
(D) = down; (U) = up
--- indicates data inclusive

TEST NUMBER 30A

RECIRCULATION SYSTEM ONE PUMP TRIP

PURPOSE

The purpose of this test is to 1) obtain recirculation system performance data during the pump trip, flow coastdown, and pump restart, and 2) verify that during the pump trip the feedwater control system can satisfactorily control water level without a resulting turbine trip or reactor scram.

CRITERIA

A. LEVEL 1

The reactor shall not scram during the one pump trip recovery.

B. LEVEL 2

The reactor water level margin to avoid a high level turbine trip shall be $\geq$ 3.0 inches during the one pump trip.

The simulated heat flux margin to avoid a scram shall be $\geq$ 5.0 percent during the one pump trip and also during the recovery.

The APRM margin to avoid a scram shall be $\geq$ 7.5% during the one pump trip recovery.

RESULTS

A. TEST CONDITION: (TC-3)

The Recirculation System One Pump Trip Test was performed during TC-3 from an initial reactor power of 75% and a core flow of 99%. The reactor water level maintained 9 inches margin to the high level turbine trip, which satisfies the Level 2 criteria. The simulated heat flux margin was 35% during the one pump trip and 24% during the one pump trip recovery, which satisfies the Level 2 criteria. The APRM margin to avoid a scram was 40% during the one pump trip recovery, which satisfies the Level 2 criteria. The one pump trip recovery was performed from 41% reactor power. The reactor did not scram during the pump trip recovery, which satisfies the Level 1 criteria. Hence, all Level 1 and 2 acceptance criteria were met.

TEST NUMBER 30B

RECIRCULATION SYSTEM TWO PUMP TRIP

PURPOSE

The purpose of this test is to record and verify acceptable performance of the recirculation two pump trip circuit system.

CRITERIA

LEVEL 1

The two pump drive flow coastdown transient during the first six seconds must be bounded by the limiting curves. (These curves are tabulated in Tables 7 and 8.)

LEVEL 2

None

RESULTS

A. TEST CONDITION: (TC-3)

The Recirculation System Two Pump Trip Test was performed during TC-3 from 73% reactor power and 95% core flow. (The test verified acceptable performance of the recirculation two pump trip circuit.) The drive flow coastdown test results are summarized for both loops in Table 7 and 8. The drive flow coastdown for loops A and B was below the upper bound (flow coastdown limit), but was not above the lower bound (ECCS analysis limit) as required. The test exception was analyzed by the General Electric Plant Transient Performance Engineering and found to be acceptable. The basis of this conclusion is an ECCS pump coastdown sensitivity study which establishes a 3.5 second coastdown minimum inertial time constant versus the curves 5 second time constant. The 3.5 second inertial time constant results in a peak clad temperature increase of less than 10°F. A 10°F peak clad temperature increase does not impact the MAPLHGR limits and therefore the measured coastdown is considered to be acceptable.

TEST NUMBER 30B (Continued)

TABLE 7

LOOP A COASTDOWN

Time After Pump Trip* (Seconds)	Drive Flow (GPM)	Drive Flow (%)	Level 1 Criteria (Lower Bound)	Level 1 Criteria (Upper Bound)	Deviation From Lower Bound Criteria (%)
0	39099	100.0	100.0	100.0	0.0
0.262	38664	98.89	99.67	99.84	-0.8
0.762	36653	93.74	96.29	97.04	-2.6
1.762	31380	80.26	85.26	87.37	-5.8
2.762	26216	67.05	74.88	78.20	-10.5
3.00	25101	64.20	72.79	76.32	-11.8
3.762	22682	58.01	66.81	70.73	-13.2
4.762	19679	50.33	59.96	64.54	-16.1
5.762	17749	45.40	55	59.5	-17.4
6.00	17341	44.35	54	58.0	-17.9

*Time zero (t=0) was adjusted by arc suppression time for experimental flow (0.02 seconds).

TEST NUMBER 30B (Continued)

TABLE 8

LOOP B COASTDOWN

Time After Pump Trip* (Seconds)	Drive Flow (GPM)	Drive Flow (%)	Level 1 Criteria (Lower Bound)	Level 1 Criteria (Upper Bound)	Deviation From Lower Bound Criteria (%)
0	39444	100.0	100.0	100.0	0.0
0.755	38607	97.87	99.11	99.34	-1.2
1.755	35433	89.83	92.15	93.34	-2.5
2.755	31598	80.11	82.22	84.93	-2.6
3.000	30694	77.82	79.81	82.80	-2.5
3.755	28263	71.65	73.07	76.55	-1.9
4.755	25333	64.23	65.31	69.59	-1.6
5.755	22551	57.17	59	64.0	-3.1
6.000	21970	55.70	58	63.5	-4.9

*Time zero (t=0) was adjusted by arc suppression time for experimental flow (0.02 seconds).

TEST NUMBER 30C

RECIRCULATION SYSTEM PERFORMANCE

PURPOSE

The purpose of this test is to record recirculation system parameters during the power test program.

CRITERIA

A. LEVEL 1 (TC-2 and TC-3)

None

B. LEVEL 2 (TC-2)

The measured recirculation pump efficiency shall not be more than eight percent below the vendor tested efficiency.

C. LEVEL 2 (TC-3)

The measured recirculation pump efficiency shall not be more than eight percent below the vendor tested efficiency.*

The jet pump nozzle and riser plugging criteria shall not be exceeded.

*The predictions are provided in General Electric document 457HA802, Rev. 1.

RESULTS

A. TEST CONDITION: TC-2

Steady state recirculation system performance data was obtained at 41% rated reactor power and 50% core flow. The recirculation pump efficiency was evaluated based on this data. Recirculation pumps 'A' and 'B' were determined to have efficiencies of 3.1% and 5.3%, respectively, below the vendor tested efficiencies. Hence, the TC-2 pump efficiency acceptance criteria of not more than eight percent below the vendor tested efficiency was met.

TEST NUMBER 30C (Continued)

B. TEST CONDITION: TC-3

Recirculation system performance data was obtained with reactor power between 20% and 75% rated, and with core flow between 27% and 108% of rated core flow. Recirculation pump 'A' and 'B' were determined to have efficiencies of 4.7% and 3.5%, respectively, below the vendor tested efficiencies. Hence, the TC-3 pump efficiency criteria of not more than eight percent below the vendor tested efficiency was met.

The jet pump nozzle and riser plugging calcualtions were performed using the General Electric jet pump calcuation computer code, JRPMP01. The jet pump nozzle plugging was determined to be 6.2% which does not exceed the the performance specification limit of 12% and the riser plugging was determined to be 5.8% which does not exceed the performance specification limit of 10%. Hence, the TC-3 jet pump plugging criteria was met.

TEST NUMBER 30D

RECIRCULATION SYSTEM FLOW CONTROL VALVE RUNBACK

PURPOSE

The purpose of this test is to verify the adequacy of the recirculation system flow control valve runback to mitigate a reactor scram upon the loss of one feedwater pump.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2

The recirculation flow control valves shall runback upon a trip of the runback circuit.

RESULTS

A. TEST CONDITION: TC-3

The Recirculation Flow Control Valve Runback Test was performed during Test Condition 3 from 68% rated reactor power and 95% rated core flow. The loss of one feed pump was simulated with a test switch, initiating a trip of the runback circuit. Flow control 'A' runback from 68% open position to 20% and flow control valve 'B' runback from 69% open position to 32%. The acceptance criterion was met.

TEST NUMBER 30E

RECIRCULATION SYSTEM CAVITATION

PURPOSE

The purpose of this test is to verify that no recirculation system cavitation will occur in the operable region of the power-flow map.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2 (TC-2 and TC-3)

Runback logic shall have settings adequate to prevent operation in areas of potential cavitation.

RESULTS

A. TEST CONDITION: TC-2

The Recirculation System Cavitation Test was performed during Test Condition 2 from 36% rated reactor power and 55% rated core flow. Reactor power was reduced along the constant 55% core flow line by inserting control rods until the low feedwater flow interlock logic was automatically actuated. Upon actuation, the recirculation pumps transferred from fast speed (60 Hz) to slow speed (LFMG sets) to prevent flow control valve cavitation. The pump transfer (runback) was verified to occur at greater than 3.93×10^6 lbs/hr feedwater flow, which prevents operation in the area of potential flow control valve cavitation. Hence, the acceptance criteria for TC-2 was met.

B. TEST CONDITION: TC-3

The Recirculation System Cavitation Test was performed during Test Condition 3 from 65% rated reactor power and 96% rated core flow. Reactor power was reduced along the 98% core flow line by inserting control rods until reactor power decreased to 44% rated. At this point, the reactor steam dome temperature to recirculation pump suction temperature differential was measured to be 9.9°F for the recirculation loop 'A' and 9.0°F for recircualtion loop 'B'. This point was 3% above the design jet pump nozzle cavitation line and is considered adequate to prevent operation in the cavitation region. Also, no recirculation pump cavitation was observed to occur. Therefore the setpoint of 9°F minimum reactor steam dome temperature to recirculation pump suction temperature differential for the recirculation loop 'A' and 'B' was installed in the recirculation pump runback logic. This satisfies the acceptance criteria for TC-3.

TEST NUMBER 31

LOSS OF TURBINE GENERATOR AND OFFSITE POWER

PURPOSE

This test determines electrical equipment and reactor system transient performance during a loss of auxiliary power.

CRITERIA

A. LEVEL 1 (TC-2)

Reactor protection system actions shall prevent violation of fuel thermal limits.

All safety systems, such as the Reactor Protection System, the 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 Low Presssure Core Spray, LPCI and ADS systems, and MSIV closure. Diesel generators shall start automatically.

B. LEVEL 2 (TC-2)

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.

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. If pressure sensors are available, they shall return to their initial state upon valve closure.

RESULTS

A. TEST CONDITION: TC-2

The Loss of Turbine Generator and Offsite Power Test was performed during Test Condition 2 from 30% rated reactor power. The Reactor Protection System actions prevented violation of fuel limits. All emergency diesel generators started automatically and satisfied the Technical Specification starting times. Proper instrument display for the reactor operators was demonstrated. No safety/relief valve opened. All Level 1 and Level 2 acceptance criteria were met.

TEST NUMBER 34

RPV INTERNALS VIBRATION

PURPOSE

The purpose of this test is to provide information needed to confirm the similarity between the reactor internals design and the prototype with respect to flow induced vibration.

CRITERIA

A. LEVEL 1 (TC-3)

The peak stress intensity may exceed 10,000 psi (single amplitude) when the component deformed in a manner corresponding to one of its normal or natural modes but the fatigue usage factor must not exceed 1.0.

B. LEVEL 2 (TC-3)

The peak stress intensity shall not exceed 10,000 (single amplitude) when the component is deformed in a manner corresponding to one of its normal or natural modes. This is the low stress limit which is suitable for sustained vibration in the reactor environment for the design life of the reactor components.

RESULTS

A. TEST CONDITION: (TC-3)

Reactor internal vibration measurement data was taken at selected test points between Test Conditions 2 and 3 and during Test Condition 3. These test points included the Single Recirculation Pump Trip Test (Test Number 30A), the Two Recirculation Pump Trip Test (Test Number 30B), and during extended core flow testing. Analysis of the vibration data determined the acceptance criteria were met.

TEST NUMBER 35

RECIRCULATION SYSTEM FLOW CALIBRATION

PURPOSE

The purpose of this test is to perform complete calibration of the installed recirculation system flow instrumentation.

CRITERIA

A. LEVEL 1 (TC-3)

None

B. LEVEL 2 (TC-3)

Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide a correct core flow indication at rated conditions.

The APRM/RBM flow-bias instrumentation shall be adjusted to function properly at rated conditions.

The flow control system shall be adjusted to limit maximum core flow to 102.5% of rated by limiting the flow control valve opening position.

RESULTS

A. TEST CONDITION: TC-3

The recirculation system flow data was taken during Test Condition 3 at 66.4% rated reactor power and 98% core flow. Based on this measured data, the General Electric jet pump calibration computer program, JRPMP01, was run to calculate the total core flow, loop flow variations, and individual jet pump riser and nozzle plugging based on elbow tap transmitter drive flow and the calibrated jet pump calculated m-ratio. All flow variation and plugging performance guidelines were satisfied. The jet pump loop flow meters and the total core flow recorder were then calibrated. This calibration satisfied the Level 2 acceptance criteria concerning jet pump flow instrumentation. The ARPM/RBM flow-biased instrumentation was respanned, satisfying Level 2 acceptance criteria concerning APRM/RBM flow-bias instrumentation. The flow control system was adjusted to limit the maximum core flow to 102.5% by limiting the flow control valve opening position, which satisfied the remaining Level 2 acceptance criteria. Hence, all acceptance criteria were met.

TEST NUMBER 72

DRYWELL ATMOSPHERE COOLING SYSTEM

PURPOSE

The purpose of this test is to verify the ability of the Drywell Atmosphere Cooling System to maintain design conditions in the drywell during operating conditions and post scram conditions.

CRITERIA

A. LEVEL 1

None

B. LEVEL 2

The drywell cooling system shall maintain an average ambient air temperature of 135°F or less and an 150°F or less ambient air temperature at any single location in containment.

RESULTS

A. TEST CONDITION: TC-2

The Drywell Atmosphere Cooling System Test was performed during Test Condition 2. The average drywell temperature was 82.4°F and the highest individual temperature recorded was 147°F near the main steam line safety relief valves. The Level 2 criteria was met.

B. TEST CONDITION: TC-3

The Drywell Atmosphere Cooling System Test was performed during Test Condition 3. A design modification was made to reduce temperatures in the RPV head area. The modification reversed the cooling air flow direction in the area between the RPV and the sacrificial shield wall. As a result of this modification some temperature limits, at the bottom of the sacrificial shield wall, were increased to 210°F by Engineering Evaluation. The average drywell temperature was 122°F and the peak temperature measured was 146°F in the upper drywell area. The Level 2 criteria was met.

TEST NUMBER 74

OFF-GAS SYSTEM

PURPOSE

The purposes of this test are to verify the proper operation of the Off-gas System over its expected operating parameters and to determine the performance of the activated carbon adsorbers.

CRITERIA

A. LEVEL 1 (TC-3)

The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications.

There shall be no loss of flow of dilution steam to the noncondensing stage when the steam jet air ejectors are pumping.

B. LEVEL 2 (TC-3)

The system flow, pressure, temperature, and relative humidity shall comply with design specifications. The catalytic recombiner, the hydrogen analyzer, the activated carbon bed, and the filters shall be performing their required function.

RESULTS

A. TEST CONDITION: TC-3

The Off-gas System was functionally tested during Test Condition 3 at 49% rated reactor power. Off-gas system data, hydrogen analyzer data, recombiner performance data, fisson product noble gas residence times data, and post filter data were collected. All off-gas system temperatures, pressures, and flows were within the operational limits for the system, except system flow, at 140 scfm, was higher than the maximum desirable flow rate of 30 scfm. Fission product noble gas residence times and post filter performance could not be evaluated at Test Condition 3 because no activity was detectable in either case. The hydrogen concentration downstream of the recombiner was $\leq$ 0.03% which is less than the maximum allowable concentration of 0.1%. The release of radioactive gaseous and particulate effluent was within the limits specified in the Technical Specifications. In order to provide the additional steam flow needed for proper off-gas dilution, piping was added to the steam jet air ejectors (SJAE's) to pass steam from the steam supply line to the second stage nozzle, around the second stage air ejector, and then into the off-gas flow. This alteration provides the necessary dilution flow without degrading the performance of the SJAE's. Off-gas system performance was acceptable and all acceptance criteria for Test Condition 3 were met.