Washington Public Power Supply System

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Docket No. 50-397

October 18, 1984

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 escalation testing to be submitted nine (9) months following initial reactor criticality. The first criticality of WNP-2 occurred on January 19, 1984 and this report is submitted pursuant to the requirement for a Startup Report.

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 prior to, or during, Test Condition No. 1. WNP-2 has completed the testing specified through Test Condition No. 2 and has met all Level 1 acceptance criteria. We are currently completing tests associated with Test Condition No. 3. The results of tests performed subsequent to Test Condition No. 1 are undergoing review and will be the subject of a future supplemental report. This report is being submitted as an interim report and future reports are expected to provide additional information concerning the report's content.

WASHINGTON NUCLEAR PLANT - UNIT 2
STARTUP REPORT

Attachment A contains the FSAR Chapter 14 test descriptions and test abstracts for Open Vessel, Heat Up and Test Condition No. 1. 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 total criteria for all test conditions. It should be noted that compliance to all criteria is not required or applicable for each test condition.

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

C M Powers for

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
Washington, DC 20555
Attn: Document Control Desk
Attachment A (36 copies)

AD Toth - NRC - Site (901A)
Attachment A (1 copy)

INTERIM P.A.T. REPORT

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 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, 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

Chemical factors defined in the Technical Specifications 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

Not applicable.

RESULTS

A. TEST CONDITION: TC-HEATUP AND TC-1

During open vessel heatup and TC-1 chemical and radiochemical tests of reactor water, condensate demineralizer inlet and effluent, feedwater, off-gas pre-treatment monitor, and post treatment monitor were conducted. Measurements of stored water (demineralized water storage, condensate storage tank, suppression pool, and condenser hotwell) and condensate and feedwater systems water quality filterable iron concentrations were taken. The results were within limits.

TEST NUMBER 2

RADIATION MEASUREMENTS

PURPOSE

The purpose of this test is 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

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

B. LEVEL 2

Not applicable.

RESULTS

A. TEST CONDITION: TC-ALL

A "complete standard survey" of background radiation levels in the plant environs was performed prior to fuel loading, during initial heatup to the rated pressure and at TC-1 (about 15% rated thermal power). All measurements indicated the radiation levels were well below all applicable criteria.

TEST NUMBER 3

FUEL LOADING

PURPOSE

The purpose of this test is to load fuel safely and efficiently to the full core size.

CRITERIA

A. LEVEL 1

The partially loaded core must be subcritical by at least 0.38% k/k with the analytically strongest rod fully withdrawn.

B. LEVEL 2

Not applicable.

RESULTS

A. TEST CONDITION: OPEN VESSEL

Fuel loading commenced with the loading of the first bundle at 0656 on 12/25/83. Loading was completed 19 days later with the seating of the last of the 764 fuel bundles at 1700 on 1/12/84. After the first 144 bundles had been loaded in a 12x12 array at the center of the core a partial core shutdown margin demonstration was successfully performed. After the core was fully loaded the seating, orientation, and location of all bundles was verified to be correct. All applicable acceptance criteria were met.

TEST NUMBER 4

FULL CORE SHUTDOWN MARGIN

PURPOSE

The purpose of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.

CRITERIA

A. LEVEL 1

The shutdown margin of the fully loaded, cold (68°F or 20°C), xenon-free core occurring at the most reactive time during the cycle must be at least 0.38% $\Delta k/k$ with the analytically strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% $\Delta k/k$ plus an exposure-dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin.

B. LEVEL 2

a. Criticality should occur within $\pm$ 1% $\Delta k/k$ of the predicted critical.

RESULTS

A. TEST CONDITION OPEN VESSEL

The full core shutdown margin with the analytically strongest rod withdrawn was determined to be 2.716% $\Delta k/k$. A 0.015% $\Delta k/k$ difference was observed between the actual and theoretical critical eigenvalues. All test criteria were satisfied.

TEST NUMBER 5

CONTROL ROD DRIVE SYSTEM

PURPOSE

The purpose of the Control Rod Drive System test is a) to demonstrate that the Control Rod Drive (CRD) System operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and b) to determine the initial operating characteristics of the entire CRD system.

CRITERIA

A. LEVEL 1

Each CRD must have a normal withdraw speed less than or equal to 3.6 inches per second (9.15 cm/sec), indicated by a full 12-foot stroke in greater than or equal to 40 seconds.

The mean scram time of all operable CRDs at any reactor pressure must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are de-energized until position 05).

Rod Position	Scram Time (Seconds)
45	0.430
39	0.868
25	1.936
05	3.497

The mean scram of the three fastest CRDs in a two-by-two array at any reactor pressure must not exceed the following times: (Scram time is measured from the time the pilot scram valve solenoids are de-energized until position 05).

Rod Position	Scram Time (Seconds)
45	0.455
39	0.920
25	2.052
05	3.706

TEST NUMBER 5 (Continued)

CRITERIA

B. LEVEL 2

Each CRD must have a normal insert or withdraw speed of 3.0 + 0.6 inches per second (7.62 + 1.52 cm/sec), indicated by a full 12-foot stroke in 40 to 60 seconds. With respect to the Control Rod Drive friction tests, if the differential pressure variation exceeds 15 psid (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.

RESULTS

A. TEST CONDITION: OPEN VESSEL

During and after Fuel Load normal insert and withdraw timing, RPIS verification, and coupling check tests were performed, along with the continuous insert friction test. After Fuel Load zero reactor pressure scram tests were conducted. The four slowest rods of both rod withdrawal sequences were also tested at a lower accumulator pressure. All acceptance criteria for these tests were satisfied. Refer to Table 1 for the mean scram time for all CRDs.

B. TEST CONDITION: HEATUP

During heatup Control Rod Drive System performance was evaluated over a range of temperatures and pressures. Heatup testing included: flow controller tuning, insert and withdrawal timing, scram timing, and friction testing. All test criteria were satisfied. Refer to Table 2 for the measured scram times.

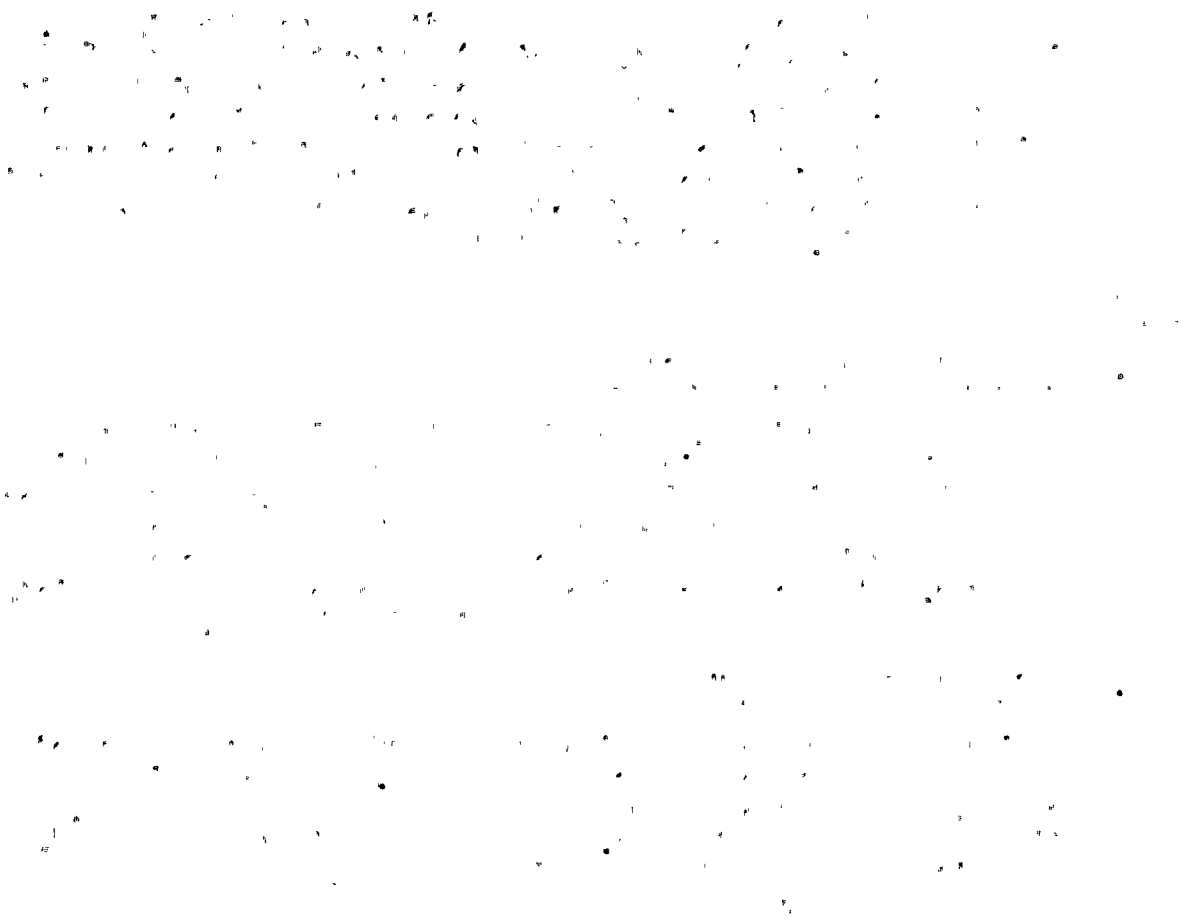


TABLE 1

MEAN SCRAM TIMES FOR ALL CRDs
at
ZERO REACTOR PRESSURE

Position Inserted From Fully Withdrawn	Mean Scram Time (Seconds)	Level 1 Criteria Value
45	0.26	0.430
39	0.44	0.868
25	0.90	1.936
05	1.64	3.497

TABLE 2

SCRAM TIMES FOR TC-HEATUP

Position Inserted From Fully Withdrawn	Scram Time of Slowest Rod (Seconds) *	Level 1 Criteria	
		Mean Scram Time (Seconds)	Mean Scram Time of 3 Fastest in 2x2 (Sec.)
45	.386	0.43	0.45
39	.733	0.86	0.92
25	1.699	1.93	2.05
05	2.923	3.49	3.70

*The time listed for each notch position is not necessarily for the same rod.

TEST NUMBER 6

SRM PERFORMANCE AND CONTROL ROD SEQUENCE

PURPOSE

The purpose of this test is to demonstrate that the operational SRM instrumentation, and rod withdrawal sequences provided adequate information to achieve criticality and increase power in a safe and efficient manner. The effect of typical rod movements on reactor power will be determined.

CRITERIA

A. LEVEL 1

There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRMs or Fuel Loading Chambers.

There must be a minimum count rate of 3 counts/second on the required operable SRMs or fuel loading chambers.

The IRMs must be on scale before the SRMs exceed the rod block setpoint.

B. LEVEL 2

Not applicable.

RESULTS

A. TEST CONDITION: OPEN VESSEL

The Source Range Monitor System performance was demonstrated during the initial critical test which was performed in conjunction with the Full Core Shutdown Margin Test and IRM performance. The SRM Test was performed for both rod withdrawal sequence "A" and sequence "B". All SRM Channels met the acceptance criteria except for channel "D" which failed during the sequence withdrawal. The problem has been traced to the SRM detector failing. The SRM "D" detector was replaced and was checked out at a subsequent critical prior to proceeding to Test Condition Heatup.

TEST NUMBER 6 (Continued)

B. TEST CONDITION: HEATUP

During the initial reactor heatup the SRMs, IRMs, and Control Rod Withdrawal Sequence were demonstrated to provide adequate information to achieve criticality, increase power, and heatup to rated temperature and pressure in a safe and efficient manner. All test criteria were satisfied.

C. TEST CONDITION: TC-1

During power ascension to Test Condition 1 (approximately 20% power). Control Rod Sequence "A" performance and core response were evaluated and deemed satisfactory. The power ascension was accomplished in a safe and efficient manner.

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 SRM and APRM systems.

CRITERIA

A. LEVEL 1

Each IRM channel must be adjusted so that overlap with the SRMs and APRMs is assured. The IRMs must produce a scram at 96% of full scale.

B. LEVEL 2

Not applicable.

RESULTS

A. TEST CONDITION: OPEN VESSEL

During the initial critical test the Intermediate Range Monitor System (IRM) was adjusted to obtain sufficient overlap with the SRM System. All applicable test criteria were satisfied.

B. TEST CONDITION: HEATUP

During Test Condition Heatup, the Intermediate Range Monitor System was tested to verify proper adjustment. The IRM System was adjusted to provide proper correlation between range 6 and 7 and sufficient overlap with the APRM System. IRM response to neutron flux was verified as well.

All applicable test criteria were satisfied.

C. TEST CONDITION: TC-1

During Test Condition 1, the IRM/APRM overlap was verified after the APRM System received its initial calibration. All acceptance criteria were met.

TEST NUMBER 11

LPRM CALIBRATION

PURPOSE

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

CRITERIA

A. LEVEL 1

Not applicable.

B. LEVEL 2

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

RESULTS

A. TEST CONDITION: HEATUP

Local Power Range Monitor (LPRM) response to flux changes and proper connection to the readout equipment was verified in conjunction with Control Rod Scram Testing at rated pressure during test condition heatup. All LPRM detectors functioned properly.

B. TEST CONDITION: TC-1

Local Power Range Monitors (LPRM) were calibrated with the aid of the off-line computer program BUCLE. 120 of 172 LPRMs failed to read within 10% of their calculated values. An additional LPRM calibration will be performed at a subsequent test condition, at which the reactor power is high enough to allow an accurate calculation of the gain adjustment factor (GAF).

TEST NUMBER 12

APRM CALIBRATION

PURPOSE

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

CRITERIA

A. LEVEL 1

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.

In the startup mode, all APRM channels must produce a scram at less than or equal to 15% of rated thermal power.

Recalibration of the APRM system will not be necessary from safety considerations if at least two APRM channels per RPS trip circuit have readings greater than or equal to core power.

B. LEVEL 2

If the above criteria are satisfied then the APRM channels will be considered to be reading accurately if they agree with the heat balance to within $\pm$ 7% of rated power.

RESULTS

A. TEST CONDITION: HEATUP

A heat balance was conducted during the initial heatup to determine thermal power and the APRMs were calibrated to the necessary accuracy of the criteria requirements.

B. TEST CONDITION: TC-1

A reactor heat balance calculation was performed at TC-1 and all APRMs were adjusted to read actual thermal power. Scram and rod block setpoints were recorded. All applicable 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

Not applicable.

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 then 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 then 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 calibration 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

A. TEST CONDITION: OPEN VESSEL

The Static System Test Case (SSTC) of NSSS and BOP software was performed after the completion of process computer preoperational test. The interface between TIP and PC was demonstrated successfully at the open vessel condition. NCCT (Number of Counts Core Top) and NCCB (Number of Counts Core Bottom) of TIP alignment data were determined and were re-verified for adequacy with the reactor at rated temperature during heatup testing.

B. TEST CONDITION: HEATUP

With the reactor at rated temperature and pressure, TIP core top (NCCT) and bottom (NCCB) limits were set to allow for the increase to rated temperature for full in maximum travel limits (Number of Counts Full In - NCFI) from their cold values for all TIP channels. A TIP flux probing monitor and XY plotter calibration was performed.

C. TEST CONDITION: TC-1

On 5/9/84, a full manual TIP set was run, all channels showed the fourth space dip 1 and 3 inches above the actual fourth spacer elevation of 79 inches above the core bottom. After adjustment of the core top and bottom limits a complete OD-1, "Whole-Core LPRM Calibration and BASE Distribution", was run. Good relative agreement was obtained between the TIP traverse data from the OD-1 edits and their corresponding TIP traces from the X-Y recorder. Programs OD-5, "Core Thermal Limits Estimate", and OD-18, "LPRM Alarm Trip Recalculation", were not tested during TC-1 testing. These programs will be verified in later test conditions.

TEST NUMBER 14

RCIC 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

The time from actuating signal to required flow must be less than 30 seconds at any reactor pressure between 150 psig (10.5/kg/cm^2) and rated.

With pump discharge at any pressure between 150 psig (10.5 kg/cm^2) and 1220 psig (85.8 kg/cm^2), the required flow is 600 gpm. (The limit of 1220 psig includes a conservatively high value of 100 psi for line losses. The measured value may be used if available.)

The RCIC turbine must not trip off during startup.

If any Level 1 criteria are not met, the reactor operation will be restricted to the power level defined by WNP-2 FSAR Figure 14.2-5. This restriction is in addition to any restrictions defined by the Technical Specification.

B. LEVEL 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 300% of the maximum required steady state flow.

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

The speed control loops shall be adjusted so that the decay ratio of any RCIC system related variable is not greater than 0.25. Note: The Flow Control Loop should be set for no observable second overshoot during CST tests. The Flow Control Loop during reactor injection tests shall be adjusted so that the decay ratio of any system related variable is not greater than 0.25.

TEST NUMBER 14 (Continued)

RESULTS

A. TEST CONDITION: HEATUP

Controlled and hot quick starts of the RCIC System injecting to the Condensate Storage Tank (CST) were performed during the heatup test phase. These tests were performed at 150 psig and at rated pressure with the pump discharge pressure throttled to 100 psi above reactor pressure. During this testing phase controller tuneup and proper system performance were verified. RCIC System control from the remote shutdown panel (RSP) was verified at rated pressure with flow recirculated back to the CST. Flow step changes were performed to tuneup and verify proper controller settings from both the Control Room and RSP.

All acceptance criteria were satisfied during this testing phase.

B. TEST CONDITION: TC-1

The TC-1 phase of the RCIC System testing involved two cold quick starts with injection to the vessel, followed by flow steps of 5% and 10% to demonstrate controller stability. The quick starts were initiated at rated pressure from the Control Room and the Remote Shutdown Panel (RSP). The 5% and 10% flow steps (initial flows of 540 gpm and 270 gpm at rated pressure) were initiated from both the Control Room and RSP. The cold quick starts satisfied the Level 1 and 2 criteria. The controller stability tests below 400 gpm did not meet the Level 2 criteria for decay ratio $<.25$.

However, the automatic function of RCIC occurs at 600 gpm. The system will automatically isolate on high reactor water level and automatically restart when obtained. If operation at less than 400 gpm continuous flow is desired, the operator can achieve the same result by manually starting and stopping the system. In all cases RCIC will automatically restart when a reactor low water level occurs. Thus the decay ratio failure at flows below 400 gpm does not adversely effect the actual RCIC system operation.

In addition, RCIC CST to CST hot quick starts at 150 psig and at rated reactor pressure were repeated to confirm the system response to the controller gain change at this test condition. All Level 1 and Level 2 criteria were satisfied.

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 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

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 (56°C).

2. The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F (28°C) of the steam dome temperature.

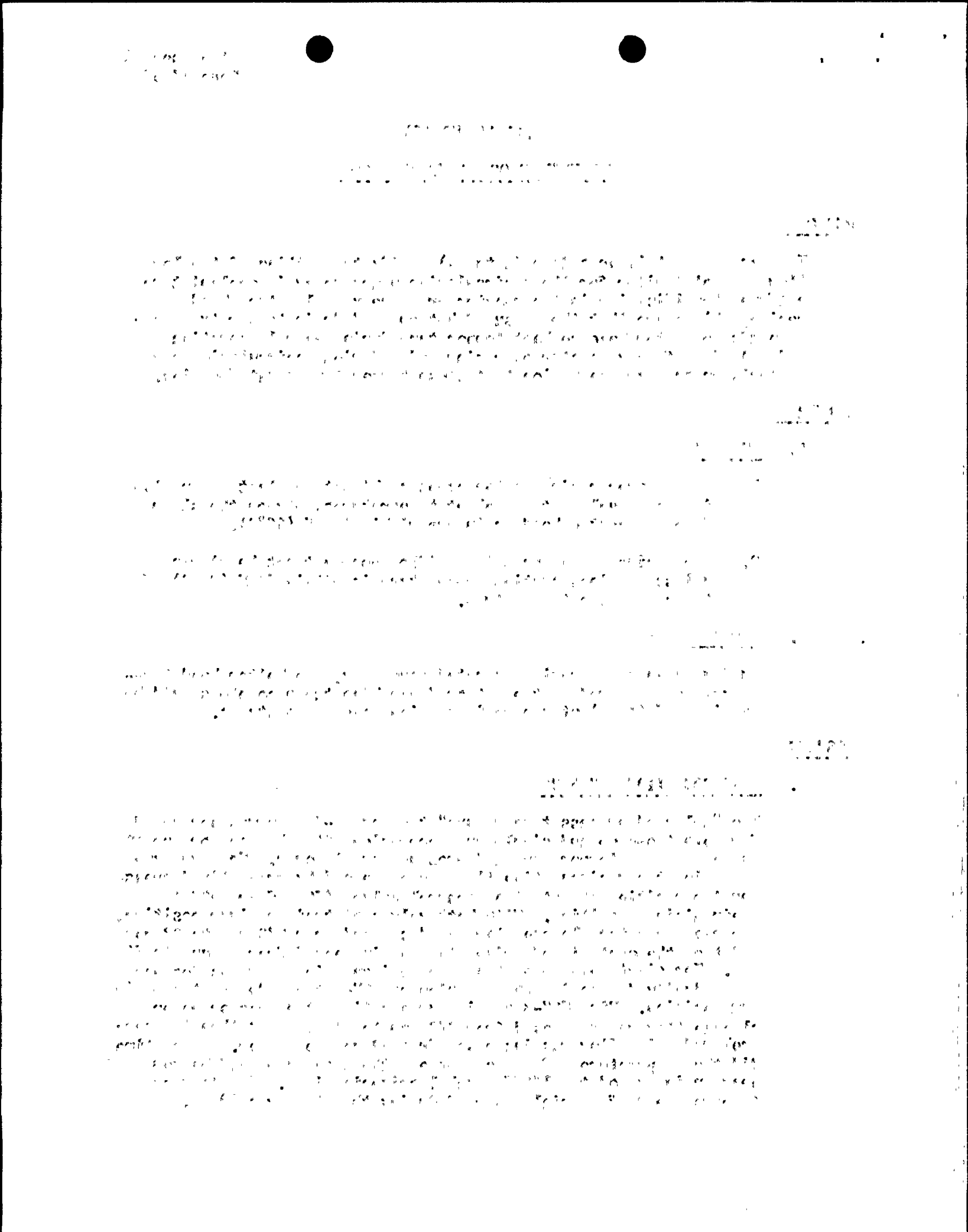
B. Level 2

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 (17°C) of the recirculation loop temperatures.

RESULTS

A. TEST CONDITION: HEATUP

The "selected process temperature" test at TC-Heatup was performed for both two-pump and single pump operation with the reactor operating near rated temperature and pressure at 4% power. For two-pump operation the recirculation flow was decreased by gradually decreasing the position of both flow control valves (FCV) from 100% to their minimum position. With both valves at their minimum position, the reactor water cleanup flow was decreased from 270 gpm to 93 gpm and then the control rod drive flow was increased from 64 gpm to 76 gpm. For single pump operation the recirculation flow was decreased by gradually decreasing the position of FCV-B from 100% to the minimum position. With FCV-B at minimum position, the reactor water cleanup flow was decreased from 270 gpm to 115 gpm and then the control rod drive flow was increased from 60 gpm to 75 gpm. At no time did the temperature difference between the steam dome and bottom head drain exceed the 145°F Level 1 criteria limit. The largest temperature differential observed during this test was 104°F.



TEST NUMBER 16A (Continued)

B. TEST CONDITION: TC-1

With the reactor in a steady state condition near rated temperature and pressure at about 17% power, data was taken to verify the absence of temprature stratification in the bottom head region. Here, the recirculation pumps were on LFMG's (Low Frequency Motor Generators) with both flow control valves A/B at 75% and 77% respectively. Minimal stratification of 38°F was observed. This was significantly less than the 145°F Level 1 criteria.

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

Not applicable.

B. LEVEL 2

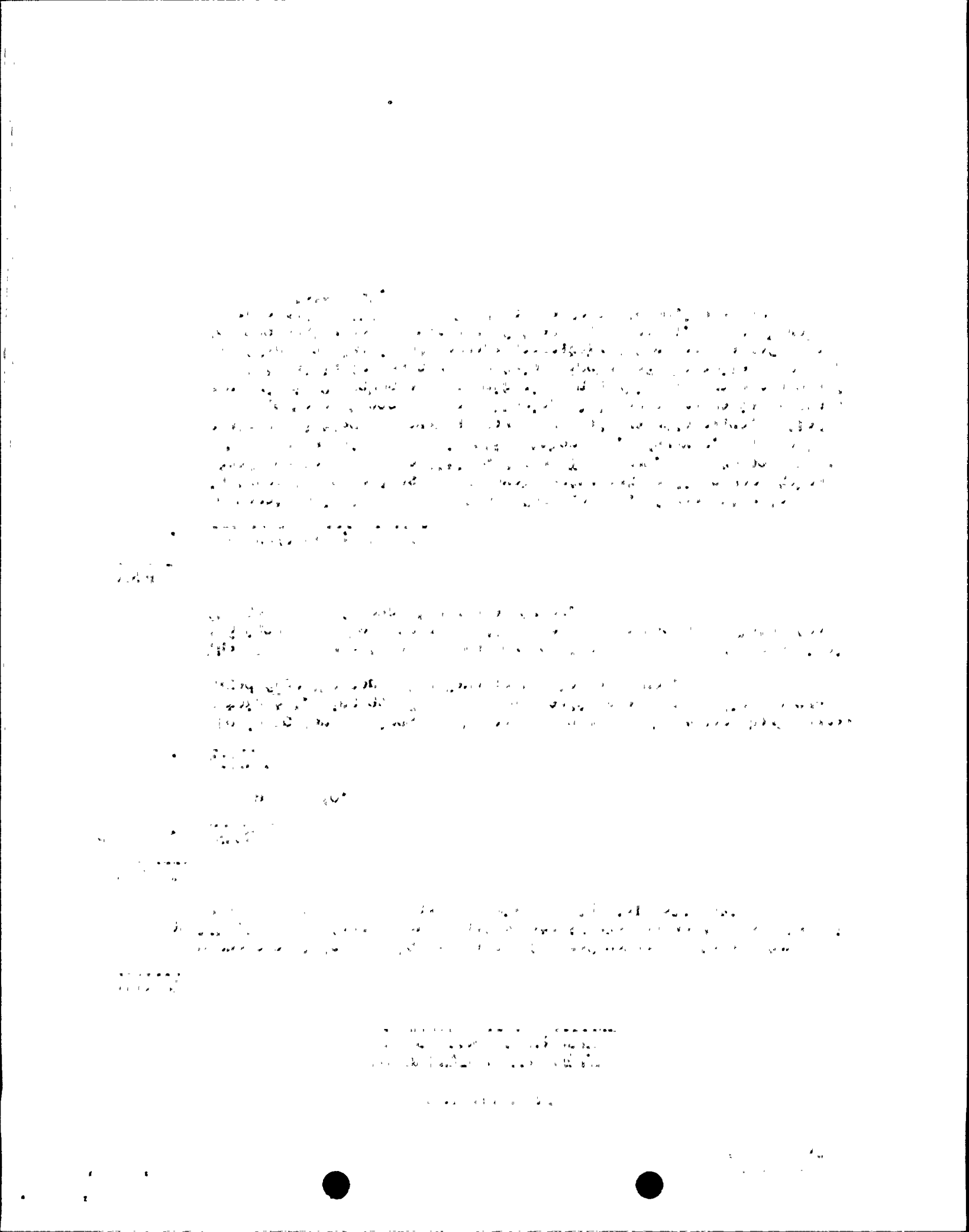
The indicator readings on the narrow range level system should agree within ± 1.5 inches of the average readings or the reading calculated from the correct reference leg temperatures.

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: HEATUP

Temperature 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 TC-Heatup. All temperatures taken were within the calibration tolerance. However, nine instruments consisting of seven narrow range and two 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 3 inches of the criteria range. An effort was initiated to more accurately determine the reference leg condensing pot elevations of these instruments. These Level 2 discrepancies will be resolved as the Power Ascension Program proceeds into TC-2.



TEST NUMBER 17

SYSTEM EXPANSION

PURPOSE

The purpose of this test is to verify that piping systems are free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. This test also verifies that all accessible snubbers installed on safety-related systems whose normal operating temperature is greater than 250°F, have adequate swing clearance to accommodate system thermal expansion.

The test also provides data for calculation of stress levels in nozzles and weldments.

CRITERIA

A. LEVEL 1

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 fully 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

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

Snubber settings must be in the operating range and should be about the midpoint of the total travel at operating temperature, or as specified on the hanger detail drawing.

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

RESULTS

A. TEST CONDITION: OPEN VESSEL

A visual inspection of the piping in the NSSS and the auxiliary systems was conducted prior to heatup. This walkdown was to verify that these selected drywell piping systems and components are unrestrained and free in regard to thermal expansion. The initial setting on the pipe supports (snubbers and hangers) and clearances on the pipe whip restraints were verified. Cables were checked to ensure they were not stretched. All the applicable criteria for the open vessel portion of this test have been satisfied.

TEST NUMBER 17 (Continued)

In addition, the remote monitor equipment (RTD's and lanyard potentiometers) which are used in later phases of this test, have been installed, calibrated, and tested. Each sensor location has been photographed and surveyed.

B. TEST CONDITION: HEATUP

Visual inspections of the drywell pipings of NSSS and the auxiliary system were conducted twice during the initial heatup. The first inspection was conducted on 11 April, 1984, with the reactor temperature at 250°F. The second inspection was conducted on 20 April, 1984, with the reactor temperature at 485°F. These walkdowns were to verify that the selected drywell piping systems and components are unrestrained and free in regard to thermal expansion. The setting on the pipe supports (snubbers and hangers) were recorded. The clearances on the pipe whip restraints were verified. Cables were checked to ensure they were not stretched. Some obstructions which could interfere with the thermal expansion of the systems inspected were found and corrected.

Displacements at the established remote monitoring equipment locations were collected through two thermal cycles. No Level 1 limits were exceeded but the measured displacement of several locations were out of the range of the Level 2 limits for the Main Steam and Recirculation Systems. This data was transmitted to G.E. Engineering for disposition of Level 2 discrepancies. These discrepancies were reveiwed and accepted by G.E. Piping Engineering. Level 2 limits for systems other than Main Steam and RRC were exceeded and evaluated by Supply System Engineering as acceptable. No Level 1 limits were exceeded on these other systems.

TEST NUMBER 19

CORE PERFORMANCE

PURPOSE

The purposes of this test are a) to evaluate the core thermal power, and b) to evaluate the following core performance parameters: 1) maximum linear heat generation rate (MLHGR), 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 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 rated MWT and values on or below the design flow control line. A core flow of 100 percent rated will not be exceeded.

B. LEVEL 2

Not applicable.

RESULTS

A. TEST CONDITION: TC-1

Reactor thermal power and core thermal limits were evaluated at TC-1. With the aid of the off-line computer program, BUCLE, Maximum Linear Heat Generation Rate (MLHGR), Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) were found to be within the limits of the acceptance criteria.

TEST NUMBER 22

PRESSURE REGULATOR

PURPOSE

The purpose of this test is a) to determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators, b) to demonstrate the automatic backup feature capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the setpoints at an appropriate value, and c) to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

CRITERIA

A. LEVEL 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to pressure regulator changes.

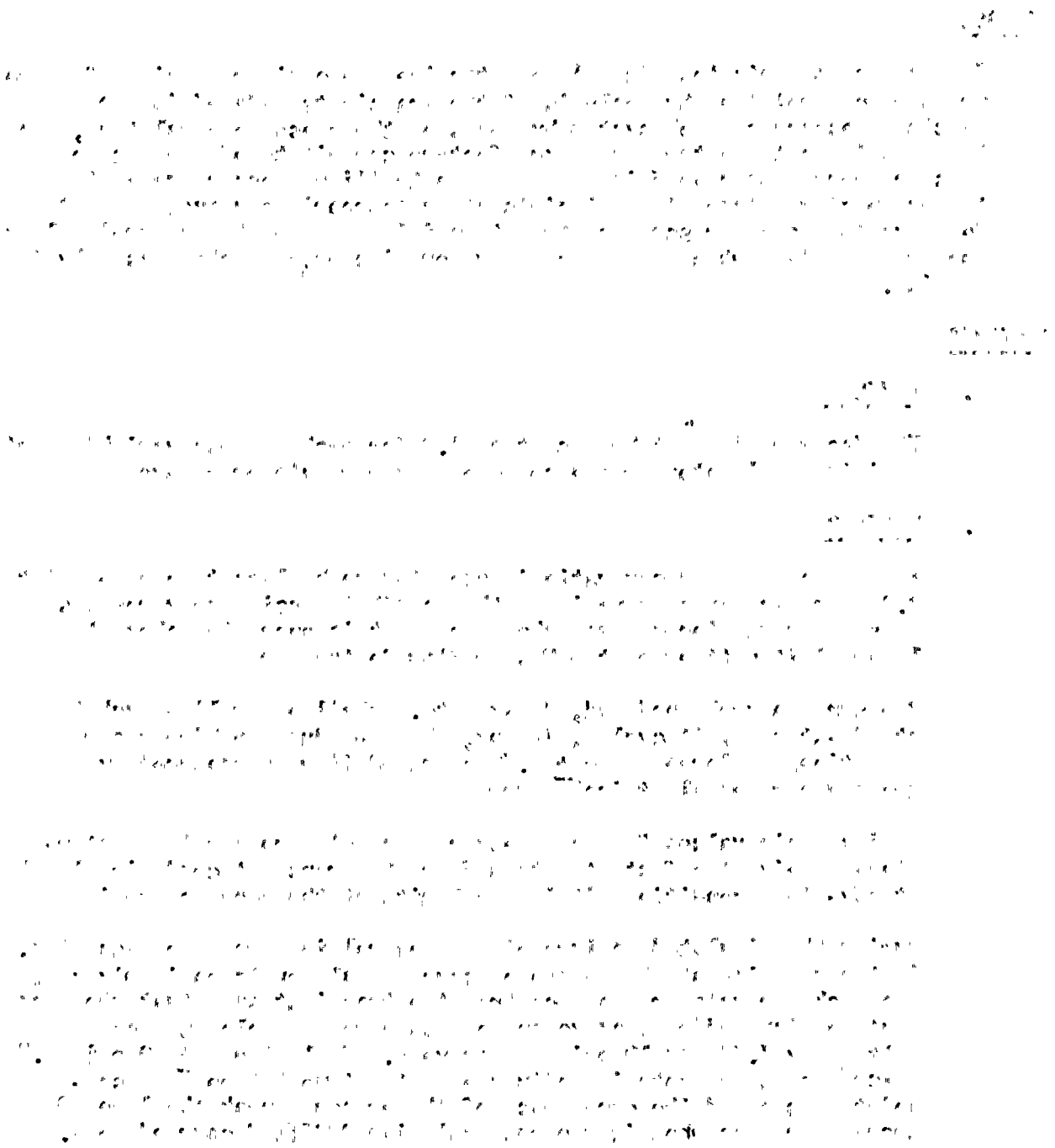
B. LEVEL 2

In all tests the decay ratio is expected to be less than or equal to 0.25 for each process variable that exhibits oscillatory response to pressure regulator changes when the plant is operating above the lower limit setting of the Master Flow Controller.

Pressure control deadband, decay, etc., shall be small enough that steady-state limit cycles, if any, shall produce turbine steam flow variations no larger than $\pm$ 0.5% of rated flow as measured by the gross generated electrical power.

Optimum gain values for the pressure control loop shall be determined to give the fastest return from the transient condition to the steady-state condition within the limits of the above criteria.

During the simulated failure of the controlling pressure regulator, if the setpoint of the backup pressure regulator is optimumly set, the backup regulator shall control the transient such that the peak neutron flux and/or peak vessel pressure remain below the scram setting by 7.5% and 10 psi respectively. Following a $\pm$ 10 psi (0.7 kg/cm^2) pressure setpoint adjustment, the time between the setpoint change and the occurrence of the pressure peak shall be 10 seconds or less when in the recirculation POSITION command mode.



TEST NUMBER 22 (Continued)

RESULTS

A. TEST CONDITION: TC-1

Pressure regulator setpoint step changes of 5, 7 and 10 psi and simulated pressure regulator failures were performed with the "A" and then the "B" regulator in control at TC-1. All test criteria, with the exception of Level 2 criterion concerning decay ratio, were satisfied. This exception was acceptable for current operating conditions. The pressure control system will be re-evaluated during subsequent test conditions with the turbine governor valves in control.

TEST NUMBER 25

MAIN STEAM ISOLATION VALVES

PURPOSE

The purpose of this test is a) to functionally check the main steam line isolation valves (MSIVs) for proper operation at selected power levels, b) to determine reactor transient behavior during the following simultaneous full closure of all MSIVs, c) to determine isolation valves closure times at rated conditions, and d) to determine 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.

Full Reactor Isolation

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIV valves 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 rated value.

Feedwater Control System settings must prevent flooding of the steam lines.

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, peak neutron flux must be 7.5% below scram, 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.

TEST NUMBER 25 (Continued)

Full Reactor Isolation

The RCIC system shall adequately maintain water level. The safety relief valves must reclose properly (without leakage) following the pressure transient.

The positive change in vessel dome pressure and simulated heat flux occurring within the first 30 seconds after the closure of all MSIV must not exceed the predicted values. These values will be referenced to actual test conditions of initial power level and dome pressure and will use BOL nuclear data. In addition, it will be corrected for the measured control rod insertion speed and the time from the start of MSIV motion to the start of control rod motion.

RESULTS

A. TEST CONDITION: HEATUP

Individual main steam isolation valve (MSIV) functional tests were performed at rated reactor temperature and pressure during TC-Heatup 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 1 for the MSIV fast closure times.

TEST NUMBER 25 (Continued)

TABLE 1

MSIV FAST CLOSURE TIMES

VALVE	CLOSURE TIME *
MS-V-22A	4.031 seconds
MS-V-22B	4.031 seconds
MS-V-22C	3.763 seconds
MS-V-22D	4.031 seconds
MS-V-28A	3.753 seconds
MS-V-28B	4.309 seconds
MS-V-28C	3.614 seconds
MS-V-28D	3.700 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 a) to verify the proper operation of the main system relief valves, b) to verify that the discharge piping is not blocked, c) to verify their proper seating following operation, d) to obtain signature information of relief valve response for subsequent comparisons, and e) to determine their capacities.

CRITERIA

A. LEVEL 1

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

The sum of capacity measurements from all relief valves shall be equal to or greater than rated, $\pm$ 2% corrected for inlet pressure of 1112 psig.

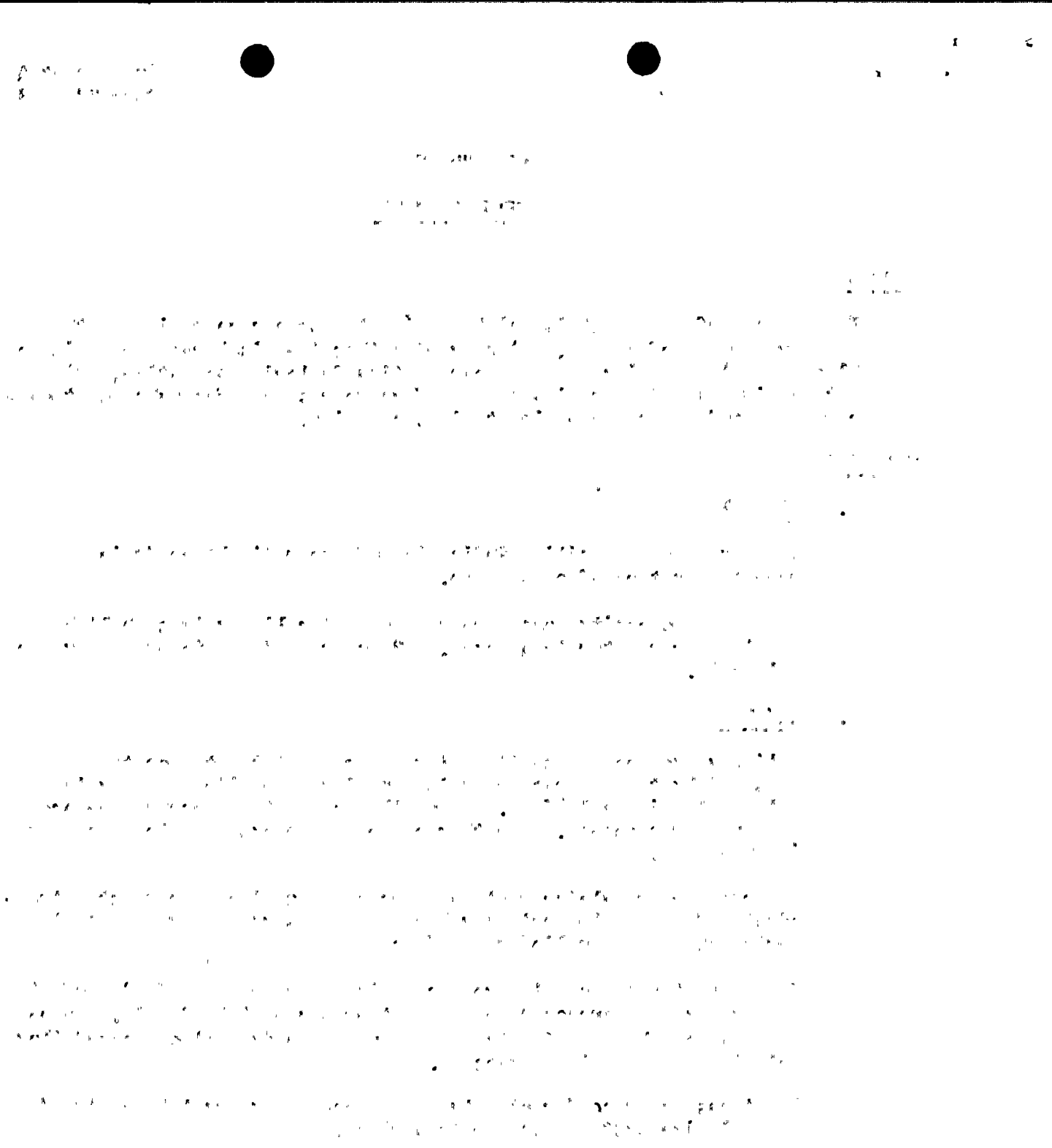
B. LEVEL 2

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 (5.6°C) of the temperature recorded before the valve was opened. The thermocouples are expected to be operating properly.

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 135% of its expected value corrected to an inlet pressure of 1112 psig. No more than 25% of the relief valves may have an individual corrected flow rate that is less than expected.

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



TEST NUMBER 26 (Continued)

RESULTS

A. TEST CONDITION: HEATUP

Proper operation of the Safety Relief Valves was verified through this test 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 pressure regulator satisfactorily controlled the reactor pressure transient during the actuation of relief valves.

Flow capacities will be determined at TC-2 or between TC-2 and TC-3.

TEST NUMBER 28

SHUTDOWN FROM OUTSIDE THE MAIN CONTROL ROOM

PURPOSE

To demonstrate that the reactor can be brought from a normal initial steady-state power level to the point where cooldown is established and under control with reactor vessel pressure and water level controlled from outside the main control room. In addition, the operation of the shutdown cooling/suppression pool cooling modes of the RHR System from the remote shutdown panel will be demonstrated.

CRITERIA

A. LEVEL 1

Not applicable.

B. LEVEL 2

During a simulated main control room evacuation, the reactor must be brought to the point where cooldown is initiated and under control. The reactor vessel pressure and water level are controlled using equipment and controls outside the main control room.

RESULTS

A. TEST CONDITION: TC-1

The reactor was satisfactorily shutdown by a team of operating personnel representing the minimum shift manning required by Plant Technical Specifications following a simulated Main Control Room evacuation. The reactor was manually scrammed and the MSIVs were closed before the control room evacuation was declared. Reactor pressure and water level were controlled from the Remote Shutdown Panel. A depressurization process was initiated to partially cool down the reactor at a rate not to exceed 100°F/hr. After the successful demonstration of this capability, control of the Plant was returned to the Main Control Room. This demonstration was done without assistance from the Main Control Room.

TEST NUMBER 29

RECIRCULATION FLOW CONTROL

PURPOSE

The purpose of this test is a) to 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) to determine that all electrical compensators and controllers are set for desired system performance and stability.

CRITERIA

A. LEVEL 1

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

B. 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 is 10 $\pm$ 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 $\leq$ 0.45 seconds for small position demand inputs of 1% to 5%. The delay time should be $\leq$ 0.15 seconds.

Flow loops are for the purpose of maintaining equal steady-state flow in two loops. Flow loop gains should be set to correct a flow unbalance in about 15 seconds.

Flux overshoot to a flux demand step shall not exceed 2% of full power.

Flux controller time constants and gain shall be adjusted to give fastest possible response within the overshoot limit given in WNP-2 FSAR 14.2.12.3.29.3.a. The response time shall be $\leq$ 2.8 seconds, when the magnitude of the demand step is within the setting of the flux error limiter. Nominal flux error setting is $\pm$ 20% of full power.

TEST NUMBER 29 (Continued)

RESULTS

A. TEST CONDITION: TC-1

The TC-1 phase of the recirculation flow control system tests verified that acceptable and conservative gain settings exist for the flow manual position control loop to support operation with the recirculation pumps operating on high speed. This was accomplished by performing a series of small step inputs (.5% to 10%) while in the POSITION command mode and operating from the low frequency motor generator (LFMG). The applicable acceptance criteria for this testing condition were satisfied.

TEST NUMBER 70

REACTOR WATER CLEANUP SYSTEM

PURPOSE

The purpose of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System. (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.)

CRITERIA

A. LEVEL 1

Not applicable.

B. LEVEL 2

The temperature at the tube side outlet of the non-regenerative heat exchangers shall not exceed 130°F in the blowdown mode and shall not exceed 120°F in the normal mode.

The pump available NPSH will be 13 feet or greater during the hot standby mode.

The cooling water supplied to the non-regenerative heat exchangers shall be within the flow and outlet temperature limits indicated in the process diagrams. (This is applicable to the "normal" and "blowdown" modes.)

RESULTS

A. TEST CONDITION: HEATUP

The Reactor Water Cleanup System (RWCU) performance was evaluated from the normal, blowdown, and hot standby modes of operation. In addition, a comparison of the bottom head flow indicator and the RWCU flow indicator was conducted. All test criteria were satisfied.

TEST NUMBER 72

DRYWELL ATMOSPHERE COOLING SYSTEM

PURPOSE

The purpose of this test is to verfy 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

Not applicable.

B. LEVEL 2

The Drywell Cooling System shall maintain drywell air temperatures and humidity at or below the design values as specified for the NSSS (SIC, Safety Related) equipment.

RESULTS

A. TEST CONDITION: TC-HEATUP AND TC-1

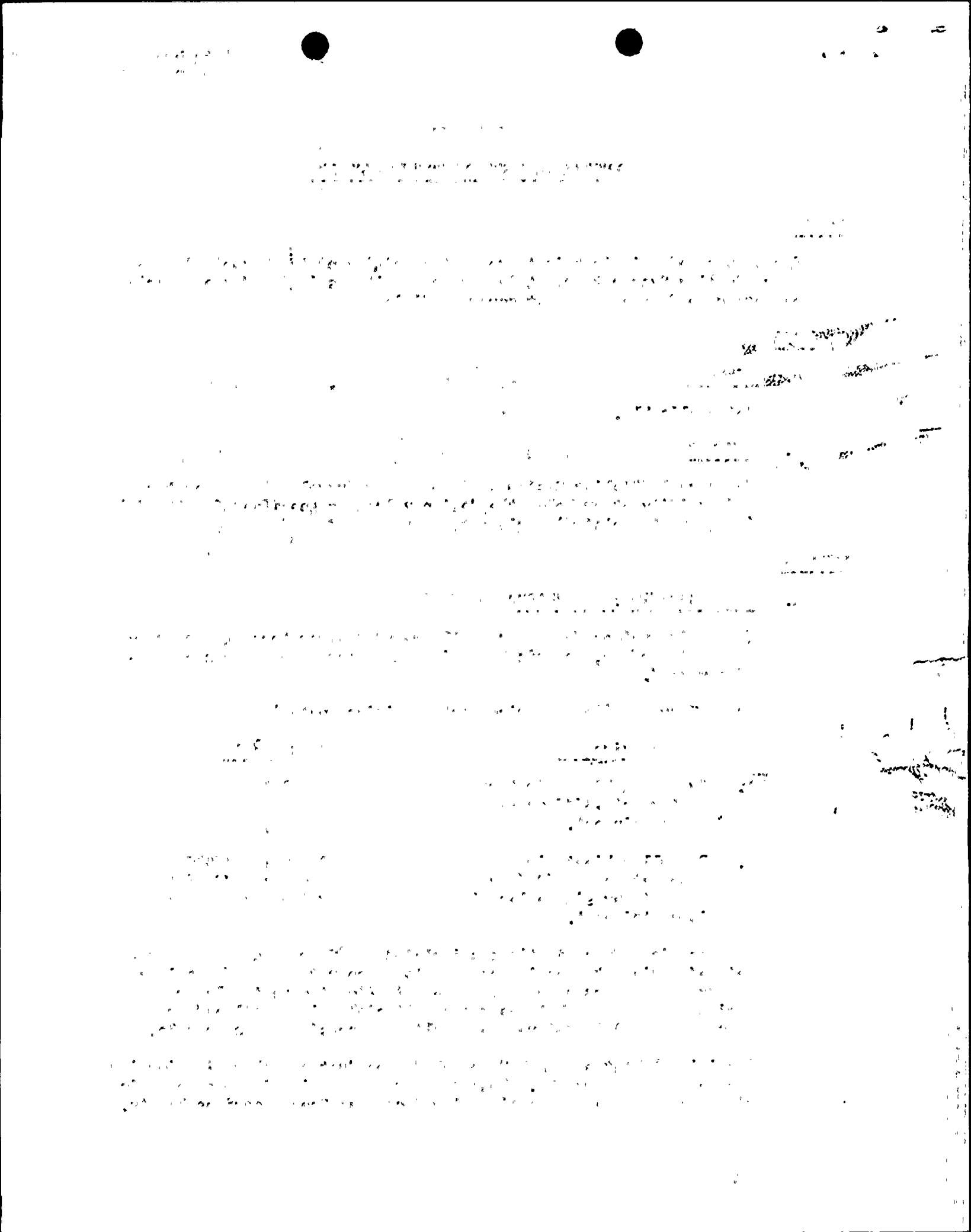
The design values for the drywell are LOCA parameters and are very large, therefore, Technical Specification parameters are conservatively used.

Temperatures: (Taken during test condition heatup)

	Requirement	Actual Value
1.	Average ambient air temperature of 135°F or less in containment.	111°F
2.	Drywell ambient air temperature of 150°F or less at any single location in containment.	All below 150°F except: CMS-TI-16 which shows 165°F

The Technical Specifications stipulated 150°F or less near safety-related equipment that is required to be operational. There is no safety-related equipment near CMS-TI-16 (located in the lower regions of the area between the sacrificial shield wall and the reactor vessel) therefore the results are considered acceptable.

The initial heatup of WNP-2 revealed some duct and flow distribution problems in the drywell. Design changes are being implemented which allow temperatures to remain below Technical Specifications limits.



TEST NUMBER 73

COOLING WATER SYSTEMS

PURPOSE

The purpose of this test is to verify that the heat removal performance of the Standby Service Water (SW) System, the Reactor Building Closed Cooling Water (RCCW) System, and the Turbine Building Service Water (TSW) System is adequate.

CRITERIA

A. LEVEL 1

Not applicable.

B. LEVEL 2

The system heat transport parameters either meet the requirements of the design specifications, or provide adequate cooling to the components serviced such that they operate satisfactorily.

RESULTS

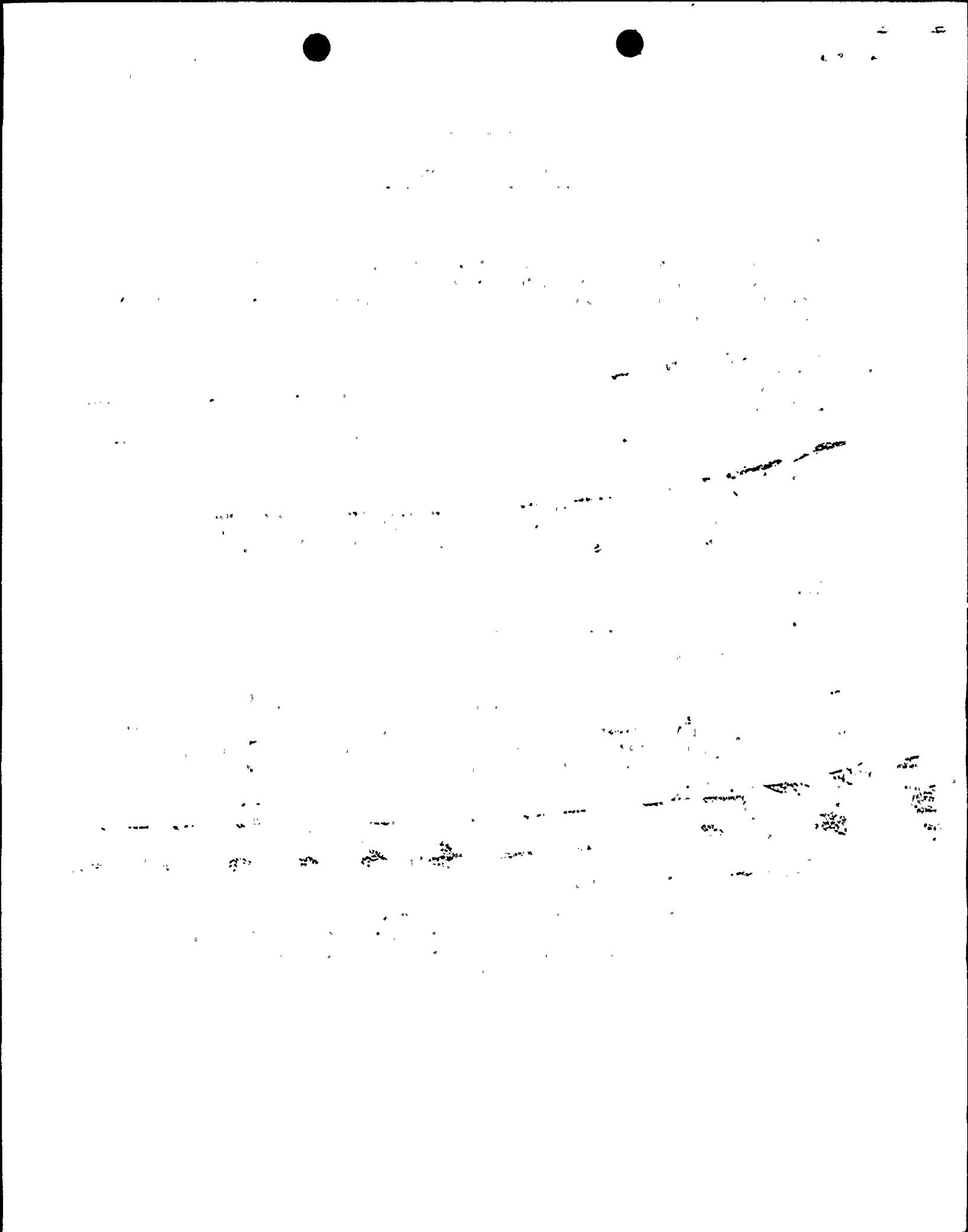
A. TEST CONDITION: TC-HEATUP AND TC-1

WNP-2 experienced the following:

Requirement	Location	Actual Value
14,366,000 BTU/hr	DG Room 1B Heat Transport	14,678,693.96 BTU/hr
7,366,000 BTU/hr	HPCS DG Area Heat Transport	5,430,000 BTU/hr
Greater than or equal to 85 ft.	HPCS-P-2 Discharge Pressure	128.47 ft

These tested components all successfully met the Level 2 requirements. The actual HPCS DG Area Heat Transport was below the requirement however, adequate cooling was verified by proper system component operation.

The majority of this test will be performed when the systems are tested in a normal controller configuration after TC-6 at design power level. This is when the maximum heat loading will be placed on the cooling water systems tested.



TEST NUMBER 74

OFF-GAS SYSTEM

PURPOSE

The purpose of this test is 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

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

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

RESULTS

A. TEST CONDITION: HEATUP

During heatup the Off-Gas System was functionally tested in the startup and normal modes of operation. System performance was judged acceptable even though the maximum dilution steam flow which could be obtained was below the suggested operational limit. System performance will be evaluated again during TC-1. All test criteria were satisfied with the exception of a few system parameters due to excessive condenser air inleakage. These parameters will be re-examined when the condenser air leakage reduction program has been completed.

B. TEST CONDTION: TC-1

The Off-Gas System was functionally tested in the normal mode of operation during TC-1. The system performance was judged acceptable, however several parameters were outside their operational range. Additional time was needed to solve the low dilution flow problem. Supply System Engineering evaluated the maximum power level obtainable, given the present steam flow, to conservatively be 40% power. Reactor power is limited to below 40% of rated until this problem can be resolved. Off-Gas System performance will be evaluated again during TC-3.