TS 6.9.1.1
TS 6.9.1.3

PHILADELPHIA ELECTRIC COMPANY

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August 30, 1991

Docket No. 50-353
License No. NPF-85

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Limerick Generating Station Unit 2
Startup Report - Cycle 2

Enclosed is the Limerick Generating Station Unit 2, Cycle 2 Startup Report. The report is being submitted in accordance with Technical Specifications Reporting Requirements 6.9.1.1 and 6.9.1.3. The report contains all pertinent information as required by TS Reporting Requirement 6.9.1.3 regarding the second cycle startup testing activities.

If you have any questions, or require additional information, please do not hesitate to contact us.

Very truly yours,

G M Leitch

KWM/cmb

Enclosure

cc: T. T. Martin, Administrator, Region I, USNRC (w/ enclosure)
T. J. Kenny, USNRC Senior Resident Inspector, LGS (w/enclosure)

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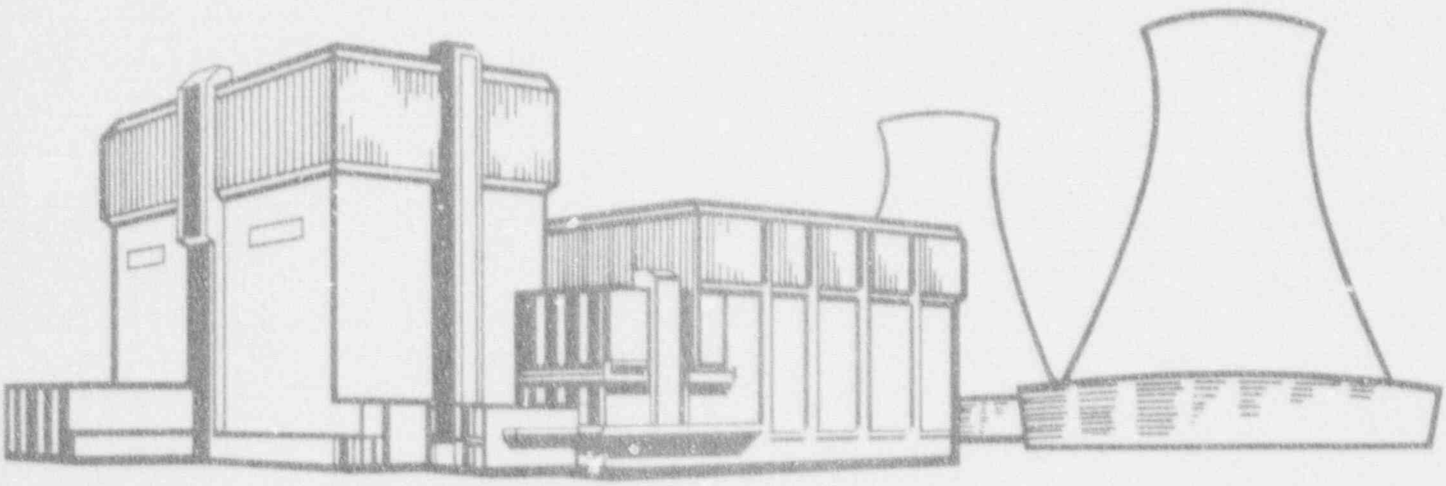

LIMERICK GENERATING STATION

August 1, 1991

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION

UNIT NO. 2

STARTUP REPORT

CYCLE 2

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INTRODUCTION/SUMMARY

1.1 REPORT ABSTRACT

This Startup Report, written to comply with Technical Specifications paragraphs 6.9.1.1 thru 6.9.1.3, consists of a summary of the Startup and Power Escalation Testing performed at Unit 2 of the Limerick Generating Station. This report is required since fuel of a different design was installed during the first refueling outage of Unit 2. During this refueling outage, 212 bundles of GE9B, 4 bundles of GE11, 4 bundles of ANF, and 4 bundles of ABB fuel were loaded into the core.

The report addresses each of the Startup Tests identified in chapter 14 of the FSAR and includes a description of the measured values of the operating conditions or characteristics obtained during the test program with a comparison of these values to the Acceptance Criteria. Also included is a description of any corrective actions required to obtain satisfactory operation.

1.2 SUMMARY

Limerick Unit 2 was out-of-service from March 22, 1991 to June 5, 1991 to accommodate a refueling outage. The unit returned to service on June 5, 1991 and reached full power operation June 10, 1991.

The successfully implemented startup program ensures that the first refuelin, outage of Limerick Unit 2 has resulted in no conditions or system characteristics that diminishes the safe operation of the plant. The tests and data referenced in this report are on file at the Limerick Generating Station.

1.3 LIMERICK PLANT DESCRIPTION

The Limerick Generating Station is a two unit nuclear power plant. The two units share a common control room, refueling floor, turbine operating deck, radwaste system, and other auxiliary systems.

The Limerick Generating Station is located on the east bank of the Schuylkill River in Limerick Township of Montgomery County, Pennsylvania, approximately 4 river miles downriver from Pottstown, 35 river miles upriver from Philadelphia, and 49 river miles above the confluence of the Schuylkill with the Delaware River. The site contains 595 acres - 423 acres in Montgomery County and 172 in Chester County.

Each of the LGS units employs a General Electric Company boiling water reactor (BWR) designed to operate at a rated core thermal power of 3293 MWt (100% steam flow) with a corresponding gross electrical output of 1092 MWe. Approximately 37 MWe are used for auxiliary power, resulting in a net electrical output of 1055 MWe. See Table 1.2-1 for Limerick Plant Parameters.

The containment for each unit is a pressure suppression type designated as Mark II. The drywell is a steel-lined concrete cone located above the steel-lined concrete cylindrical pressure suppression chamber. The drywell and suppression chamber are separated by a concrete diaphragm slab which also serves to strengthen the entire system.

The Architect Engineer and Constructor was Bechtel Power Corporation.

The plant is owned and operated by the Philadelphia Electric Company.

TABLE 1.3-1
Limerick 2 Plant Parameters

Parameter	Value
Rated Power (MWt)	3293
Rated Core Flow (Mlb/hr)	100 (1)
Reactor Dome Pressure (psia)	1020
Rated Feedwater Temperature (Deg. F)	420 (2)
Total Steam Flow (Mlb/hr)	14.159
Vessel Diameter (in)	251
Total Number of Jet Pumps	20
Core Operating Strategy	Control Cell Core
Number of Control Rods	185
Number of Fuel Bundles	764
Fuel Type	8 x 8 (Barrier) (3)
Core Active Fuel Length (in)	150
Cladding Thickness (in)	0.032 (4)
Channel Thickness (in)	0.080
MCPR Operating Limit	1.30 for GE7B and GE9B (5) 1.35 for GE11 1.47 for ABB and ANF
Maximum LHGR (KW/ft)	13.4 for GE7B 14.4 for GE9B, GE11, ABB and ANF
Turbine Control Valve Mode	Modified Partial Arc
Turbine Bypass Valve Capacity (% NBR)	25
Relief Valve Capacity (% NBR)	87.4
Number of Relief Valves	14
Recirculation Flow Control Mode	Variable Speed M/G Sets

NOTES FOR TABLE 1.3-1

(1) Unit 2 is analyzed for increased core flow to 105%.

(2) Unit 2 is analyzed for a 60 degrees F final Feedwater temperature reduction.

(3) Except for: 4 GE11 fuel bundles which are 9x9
4 ANF fuel bundles which are 9x9
4 ABB fuel bundles which are 10x10

(4) Except for: 4 GE11 fuel bundles which are 0.028
4 ANF fuel bundles which are 0.030
4 ABB fuel bundles which are 0.0248

(5) See Core Operating Limits Report for LGS Unit 2 Reload 1, Cycle 2 for specifies.

SECTION 2

RESULTS

2.1 2STP-1, CHEMICAL AND RADIOCHEMICAL

OBJECTIVES

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 analytical techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the entire reactor system meet specifications and process requirements.

Specific objectives of the test program include evaluation of fuel performance, evaluation of demineralizer operations by direct and indirect methods, measurements of filter performance, confirmation of condenser integrity, demonstration of proper steam separator-dryer operation, 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 special chemical tests.

ACCEPTANCE CRITERIA

Level 1

Chemical factors defined in the Technical Specifications must be maintained within the limits (chemical values and time intervals) specified.

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

Level 2

Chemical factors in the Fuel Warranty must be maintained within the specified limits.

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

RESULTS

During Startup of Limerick Generating Station Unit 2 reactor, following its first refueling outage, reactor coolant chemistry parameters as well as radioactive gaseous waste releases and radioactive liquid waste releases were maintained within the limits set forth in the Limerick Generating Station Unit 2 Technical Specifications. The following is a list of Chemistry related surveillance tests satisfactorily performed in support of unit startup activities:

ST-5-041-800-2, ST-5-041-875-2, ST-5-041-876-2,
ST-5-041-877-2, ST-5-041-878-2, ST-5-041-879-2,
ST-5-041-885-2, ST-5-061-570-0, ST-5-070-885-2,
ST-5-076-810-2, ST-5-076-815-0, ST-5-076-820-0

In addition to the surveillance tests, routine tests and normal analyses were performed. Maximum Dose Equivalent Iodine was 6.91 N5 uCi/g (Tech Spec 0.2 uCi/g). Fuel Warranty Appendix I - Water Quality Requirements were met during startup. From 6/2/91 through 6/16/91 with reactor power greater than 0%, reactor water conductivity averaged 0.199 umho/cm, (Fuel Warranty limit 1.0) chlorides ranged from less than 2 to 10.7 ppb (Fuel Warranty limit 100 ppb), and pH ranged from 6.21 to 7.93 (Fuel Warranty Range 5.6 - 8.6). Above 50% power, feedwater copper concentration reached a maximum of 0.27 ppb, iron reached .318 ppb and total metals reached .914 ppb (Fuel Warranty limit 2 ppb, 10 ppb, and 15 ppb respectively). The highest condensate demineralizer effluent conductivity above 50% power was 0.068 umho/cm.

Condensate and reactor water cleanup demineralizer performance was monitored closely during the startup. Demineralizers were regenerated as necessary to maintain reactor water conductivity less than 0.3 umho/cm.

2.2 2STP-2, RADIATION MEASUREMENTS

OBJECTIVES

The objectives of this test are to a) determine the background radiation levels in the plant environs prior to operation for base data to assess future activity buildup and b) monitor radiation at selected power levels to assure the protection of personnel during plant operation, and c) verify that general area dose rates and shield walls satisfy radiation zoning criteria.

ACCEPTANCE CRITERIA

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 as outlined in 10CFR20 "Standards for Protection Against Radiation".

Level 2

None

RESULTS

Health Physics procedure HP-203, "HP Startup Surveillance Procedure" was implemented during reactor startup. This procedure directs Health Physics surveillance throughout the plant to help ensure plant posting and RWP's are updated as reactor power increases.

2.3 2STP-3, FUEL LOADING

OBJECTIVE

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

ACCEPTANCE CRITERIA

Level 1

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

Level 2

None

RESULTS

The beginning of cycle shutdown margin calculated in the Cycle Management Report Limerick 2 Cycle 2 was 2.49% delta K/K. Core reload was conducted in accordance with Technical Specifications. Equipment required to be operable to ensure that the shutdown margin is maintained was verified operable by various performances of ST-6-107-630-2 and ST-6-107-591-2 between April 22, 1991 and May 1, 1991. Post alteration core verification was completed on May 1, 1991 after all refueling operations were completed by the performance of ST-3-097-355-2. All fuel bundles were verified to be in their proper core locations and properly oriented in the control cell. The bundle seating pass identified one fuel bundle improperly seated (05-50). The bundle was properly reseated, and the location and orientation was reverified after reseating.

2.4 2STP-4, SHUTDOWN MARGIN DEMONSTRATION

OBJECTIVES

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

ACCEPTANCE CRITERIA

Level 1

The shutdown margin (SDM) of the fully loaded, cold (68 degrees F), 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 it's reactivity equivalent) withdrawn. If the SDM is measured at sometime during the cycle other than the most reactive time, compliance with the above criteria is shown by demonstrating that the SDM is 0.38% delta K/K plus an exposure dependent correction factor which corrects the SDM at that time to the minimum SDM.

Level 2

Criticality should occur within ±1.0% delta K/K of the predicted critical.

RESULTS

An "In Sequence" shutdown margin of at least 1.855% delta K/K was obtained during the reactor startup. This satisfies the Level 1 acceptance criteria. Test data is documented in ST-6-107-875-2 completed on June 2, 1991.

Using the data obtained during the shutdown margin demonstration, the difference between criticality and predicted critical was 0.035% delta K/K. This was within the Level 2 acceptance criteria.

2.5 STP-5, CONTROL ROD DRIVE SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate that the Control Rod Drive (CRD) System operates properly over the full range of primary coolant operating temperatures and pressures, and to determine the initial operating characteristics of the CRD system.

ACCEPTANCE CRITERIA

Level 1

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

The mean scram time of all operable CRD's must not exceed the following times (Scram time is measured from the time the pilot scram valve solenoids are de-energized):

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

The mean scram time of the three fastest CRD's in a two by two array must not exceed the following times (Scram time is measured from the time the pilot scram valve solenoids are de-energized):

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

Level 2

Each CRD must have normal insert and withdrawn speeds of 3.0 ± 0.6 inches per second, 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 (dp) variation exceeds 15 psid 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 nor should it vary by more than 10 psid over a full stroke.

RESULTS

Although the performance of the Control Rod Drive System was not affected by the installation of the new fuel design, the scram time limits are required to assure thermal limits such as critical power ratio are not exceeded.

Level 1 and Level 2 stroke time acceptance criteria were fully satisfied by the performance of ST-6-047-760-2 on May 15, 1991.

Level 1 scram time acceptance criteria were fully satisfied by the performance of ST-3-107-790-2 on May 18, 1991 during the operational hydrostatic test.

2.6 2STP-6, SRM PERFORMANCE AND CONTROL ROD SEQUENCE

OBJECTIVES

The objective of this test is to demonstrate that the operational neutron sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner.

ACCEPTANCE CRITERIA

Level 1

There must be a neutron signal to noise count ratio of a least 2:1 on the required operable SRMs.

There must be a minimum of .7 counts per second provided that the signal to noise ratio is at least 2:1.

Level 2

None

RESULTS

Minimum SRM count rate was determined to be greater than 3 CPS by the performance of ST-6-107-591-2 prior to the withdrawal of control rods on June 2, 1991. The signal-to-noise ratio verification is only required to be performed in accordance with Tech Specs if the SRM count rate is less than 3.0 CPS.

Since at no time during the startup was the count rate less than 3.0 CPS, this verification was not performed. SRM response was verified by the performance of ST-6-107-875-2 on June 2, 1991 until criticality was achieved.

2.7 2STP-9, WATER LEVEL REFERENCE LEG TEMPERATURE

OBJECTIVES

The objectives of this test are to measure the level instrumentation reference leg temperature, recalibrate the water level instruments if the measured temperature is significantly different from the value assumed during the initial end points calibration, and to obtain baseline data on the Narrow Range and Wide Range water level instrumentation.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

The difference between the actual reference leg temperature(s) and the value(s) assumed during initial calibration shall be less than that amount which will result in a scale end point error of 1% of the instrument span for each range.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.8 2STP-10, IRM PERFORMANCE

OBJECTIVES

The objectives of this test are to adjust the Intermediate Range Monitoring (IRM) System to obtain an optimum overlap with the SRM and APRM systems.

ACCEPTANCE CRITERIA

Level 1

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

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

Level 2

Each IRM channel must be adjusted so that one-half decade overlap with the SRM's is assured.

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

RESULTS

Technical Specification SRM/IRM overlap was satisfied by the performance of ST-6-107-884-2 on June 2, 1991. This test demonstrated at least a half decade SRM/IRM overlap.

During startup, all required APRM's were verified to be on scale before any IRM exceeded their scram setpoint of 120% of scale. This was documented on GP-2, Normal Plant Startup, on June 4, 1991. One-half decade IRM/APRM overlap is verified in accordance with Technical Specifications during each controlled shutdown by the performance of ST-6-107-886-2.

2.9 2STP-11, LPRM CALIBRATION

OBJECTIVES

The objectives of this test are to calibrate the Local Power Range Monitoring (LPRM) System and to verify LPRM Flux Response.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

RESULTS

LPRM calibrations were performed at 25% power and 100% power per ST-3-074-505-2 on June 8, 1991 and June 18, 1991 respectively. On June 18, 1991, the LPRM's were calibrated to within 4% of their calculated value.

2.10 2STP-12, APRM CALIBRATION

OBJECTIVES

The objective of this test is to calibrate the Average Power Range Monitor (APRM) System.

ACCEPTANCE CRITERIA

Level 1

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

Technical specification and fuel warranty 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.

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 or the minimum value required based on peaking factor, MLHGR, and fraction of rated power to within (+7,-0)% of rated power.

RESULTS

By various performances of ST-6-107-885-2 from June 8, 1991 to June 12, 1991, Level 1 acceptance criteria was met by verifying APRM channels were indicating greater than or equal to actual core thermal power and below the scram and rod block setpoints when thermal power was greater than 25%. Level 2 acceptance criteria was also met in this surveillance test by adjusting indicated APRM reading to within +2, -0% (not to exceed 100%) of the greater of fraction of rated power or maximum fraction limiting power density.

The Level 1 acceptance criteria of APRM scram setpoint of 15% was met by performance of channel functional tests ST-2-074-412-2, ST-2-074-413-2, ST-2-074-414-2, ST-2-074-415-2, ST-2-074-416-2, and ST-2-074-417-2 performed on May 27, 1991 through June 5, 1991.

2.11 2STP-13, PROCESS COMPUTER

OBJECTIVES

The objective of this test is to verify the performance of the Process Computer under plant operating conditions.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

The Minimum Critical Power Ration (MCPR) calculated by BUCLE (the off-site Mark III computer system program) and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2% or
- for the case in which the MCPR calculated by the Process Computer is in a different assembly than that calculated by BUCLE, of each assembly, the MCPR and the CPR calculated by the two methods shall agree within 2%.

The maximum Linear Heat Generation Rate (LHGR) calculated by BUCLE and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2%, or
- for the case in which the maximum LHGR calculated by the Process Computer is in a different assembly than that calculated by BUCLE, of each assembly, the maximum LHGR and the LHGR calculated by the two methods shall agree within 2%.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) calculated by BUCLE and by PMS either:

- are in the same fuel assembly and do not differ in value by more than 2%, or
- for the case in which the MAPLHGR calculated by the Process Computer is in a different assembly than th[illegible] calculated by BUCLE, of each assembly, the MAPLHGR [illegible]d APLHGR calculated by the two methods shall agree within 2%.

The Local Power Range Monitor (LPRM) gain adjustment factors calculated by BUCLE and PMS agree to within 2%. Each LPRM reading will be within 10% of its calculated value.

RESULTS

On June 14, 1991 at 99.6% core thermal power, the accuracy of the thermal limits and LPRM gain adjustment factor calculated by the Process Computer were compared to the values calculated by an offline computer program called Backup Core Limits Evaluation (BUCLE). The acceptance criteria for thermal limits determination was satisfied in all cases. Table 2.11-1 summarizes the thermal limits data. Also, all LPRM gain adjustment factors calculated by BUCLE and the Process Computer for operable LPRM's were determined to be within 2%.

TABLE 2.11-1
LGS 2 BOC 2 100% Power P1 to BUCLE Comparison

Value	P1 Data 6/14/91, 0934	BUCLE Data 6/14/91, 0934
CMWT	3279	3279

MFLPD

Location	Value P1	Value BUCLE
11-20-5	.899	0.900
25-10-5	.928	0.929
49-20-5	.899	0.900
09-26-5	.945	0.946
33-34-5	.808	0.808
51-26-5	.945	0.946
11-42-5	.899	0.900
25-52-5	.928	0.929
49-42-5	.899	0.900

MFLCPR

Location	Value P1	Value BUCLE
11-22	.787	.787
27-06	.826	.826
49-22	.787	.787
05-28	.816	.817
29-24	.740	.740
55-28	.816	.817
11-40	.787	.787
27-56	.826	.827
49-40	.787	.787

MAPRAT

Location	Value P1	Value BUCLE
11-20-5	.897	.897
25-10-5	.925	.926
49-20-5	.897	.897
09-26-5	.943	.943
31-26-5	.796	.797
51-26-5	.943	.943
11-42-5	.897	.897
25-52-5	.925	.926
49-42-5	.897	.897

2.12 2STP-14, RCIC SYSTEM

OBJECTIVES

The objectives of this test are to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) System over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at power conditions.

ACCEPTANCE CRITERIA

Level 1

The average pump discharge flow must be equal to or greater than 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig and rated.

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

Level 2

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

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

The delta P switches of the RCIC steam supply line high flow isolation trip shall be calibrated to actuate at the value specified in the plant technical specifications (about 300%).

The RCIC system must have the capability to deliver specified flow against normal rated reactor pressure without the normal AC site power supply.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.13 2STP-15, HPCI SYSTEM

OBJECTIVES

The objectives of this test are to verify the proper operation of the High Pressure Coolant Injection (HPCI) System over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at rated pressure conditions.

ACCEPTANCE CRITERIA

Level 1

The average pump discharge flow must be equal to or greater than 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 200 psig and rated.

The HPCI turbine shall not trip or isolate during auto or manual start tests.

Level 2

In order to provide an overspeed isolation trip margin, the transient first peak shall not come closer than 15% (of rated speed) to the overspeed trip, and subsequent speed peaks shall not be greater than 5% above the rated turbine speed.

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

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

The delta P switches of the HPCI steam supply line high flow isolation trip shall be calibrated to actuate at the value specified in plant technical specifications (about 300%).

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.14 2STP-16, SELECTED PROCESS TEMPERATURES

OBJECTIVES

The objectives of this Subtest is to assure that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.15 2STP-17, SYSTEM EXPANSION

OBJECTIVES

This Subtest verifies that safety related piping systems and other piping systems as identified in the FSAR expand in the acceptable manner during plant heatup and power escalation. Specific objectives are to verify that:

Piping thermal expansion is as predicted by design calculations.

Snubbers and spring hangers remain within operating travel ranges at various piping temperatures

Piping is free to expand without interferences.

ACCEPTANCE CRITERIA

Level 1

There shall be no obstructions which will interfere with the thermal expansion of the Main Steam (inside drywell) and Reactor Recirculation piping systems.

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

Level 2

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

Snubbers and spring hangers do not become extended or compressed beyond allowable travel limits (working range) and snubbers retain swing clearance.

Measured displacements compared with the calculated displacements are within the specified range.

Residual displacements measured following system return to ambient temperature do not exceed the greater of $\pm$ 1/16 in. or $\pm$ 25% of the maximum displacements measured during system initial heatup.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.16 2STP-18, TIP UNCERTAINTY

OBJECTIVES

The objective of this test is to determine the reproducibility of the Traversing Incore Probe (TIP) system readings.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

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

RESULTS

Total TIP uncertainty was determined by the performance of RT-3-074-850-0, Core Power Symmetry and TIP Reproducibility Test on June 25, 1991. Level 2 acceptance criteria was met by all data sets with a total uncertainty of 1.41%.

2.17 2STP-19, CORE PERFORMANCE

OBJECTIVES

The objectives of this test are to:

a) Evaluate the core thermal power and core flow rate; and

b) Evaluate whether the following core performance parameters are within limits:

- Maximum Linear Heat Generation Rate (MLHGR),
- Minimum Critical Power Ratio (MCPR),
- Maximum Average Planar Linear Heat Generation Rate (MAPLHGR).

ACCEPTANCE CRITERIA

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 exceed the minimum limit specified by the Plant Technical Specifications.

The Maximum Average 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 core thermal power (3293 MWt).

Core flow shall not exceed its rated value (105 Mlb/hr).

Level 2

None

RESULTS

With thermal power limited to 3293 MWth and core flow limited to 105 Mlb/hr, Level 1 acceptance criteria of thermal limits were met and documented throughout the startup by various performances of ST-6-107-885-2 from June 8, 1991 through June 12, 1991.

2.18 2STP-20, STEAM PRODUCTION

OBJECTIVES

The objectives of this test are to demonstrate that the Nuclear Steam Supply System (NSSS) can provide steam sufficient to satisfy all appropriate warranties as defined in the NSSS contract.

ACCEPTANCE CRITERIA

Level 1

The NSSS parameters as determined by using normal operating procedures shall be within the appropriate license restrictions.

The NSSS shall be capable of supplying 14,159,000 pounds per hour of steam of not less than 99.7% quality at a pressure of 985 psia at the discharge of the second main steam isolation valve, as based upon a final reactor feedwater temperature of 420 degrees F and a control rod drive feed flow of 32,000 pounds per hour at 80 degrees F. The reactor feedwater flow must equal the steam flow less the control rod drive feed flow.

Level 2

None

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.19 2STP-22, PRESSURE REGULATOR

OBJECTIVES

The objectives of this test are as follows:

To demonstrate optimized controller settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the set point changes to the pressure regulators.

To demonstrate the take-over capability of the back-up pressure regulator upon failure of the controlling pressure regulator, and to set spacing between the setpoints at an appropriate value.

To demonstrate smooth pressure control transition between the turbine control valves and the bypass valves when reactor steam generation exceeds the steam flow used by the turbine.

ACCEPTANCE CRITERIA

Level 1

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

Level 2

Pressure control system related variables may contain oscillatory modes of response. In these cases, the decay ratio of each controlled mode of response must be less than or equal to 0.25. (This criterion does not apply to tests involving simulated failure of one regulator with the backup regulator taking over.)

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

Pressure control system deadband, delay, 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.

The peak neutron flux and/or peak vessel pressure shall remain below the scram settings by 7.5 percent and 10 psi respectively of all pressure regulator transients performed at Test Condition 6.

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", of each flow range) shall meet the following:

% of Steam Flow Obtained With Valves Wide Open	Variation
0 to 85%	$\leq$4:1
85% to 97%	$\leq$2:1
97% to 99%	$\leq$5:1

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.20 2STP-23, FEEDWATER SYSTEM

OBJECTIVES

The objectives of this test are:

To demonstrate that the feedwater system has been adjusted to provide acceptable reactor water level control.

To demonstrate an adequate response to a feedwater temperature reduction.

To demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump at high power operation.

To demonstrate that the maximum feedwater runout capability is compatible with the licensing assumptions.

ACCEPTANCE CRITERIA

Level 1

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

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be ≤100 deg. F. The resultant MCPR must be greater than the fuel thermal safety limit.

The increase in simulated heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature changes and initial power level.

Maximum speed attained shall not exceed the speeds which will give the following flows with the normal complement of pumps operating.

127% NBR at 1020 psig.

Level 2

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

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

a. Maximum time to 10% of a step disturbance $\leq$1.1 sec

b. Maximum time of 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

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

As steady-state generation of the 3/1 element systems, the input scaling to the mismatch gain should be adjusted such that the level error due to biased mismatch gain output should be within $\pm$1 inch.

The increase in simulated heat flux cannot exceed the predicted value referenced to the actual feedwater temperature change and initial power level.

The reactor shall avoid low water level scram by three inches margin from an initial water level halfway between the high and low level alarm setpoints.

The maximum speed must be greater than the calculated speeds required to supply:

a. With rated complement of pumps - 115% NBR at 1075 psia

b. One feedwater pump tripped conditions - 68% NBR at 1025 psia.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.21 2STP-24, TURBINE VALVE SURVEILLANCE

OBJECTIVES

The objectives of this test are to demonstrate acceptable procedures and maximum power levels of periodic surveillance testing of the main turbine control, stop and bypass valves without producing a reaccor scram.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

Peak neutron flux must be at least 7.5% below the scram trip setting.

Peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

Peak steam flow in each line must remain 10% below the high flow isolation trip setting.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.22 2STP-25, MAIN STEAM ISOLATION VALVES

OBJECTIVES

The objectives of this test are to functionally check the Main Steam Isolation Valves (MSIV's) of proper operation at selected power levels, to determine the MSIV closure times, and to determine the maximum power level at which full closure of a single MSIV can be performed without causing a reactor scram.

The full isolation is performed to determine the reactor transient behavior that results from the simultaneous full closure of all MSIV's at a high power level.

ACCEPTANCE CRITERIA

Level 1

MSIV stroke time shall be no faster than 3.0 seconds. MSIV closure time shall be no slower than 5.0 seconds.

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIV's 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.

Reactor must scram to limit the severity of the neutron flux and simulated heat flux transients.

Level 2

The reactor shall not scram. The peak neutron flux must be at least 7.5 percent below the trip setting. The peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

The reactor shall not isolate. The peak steam flow on each line must remain 10 percent below the high steam flow isolation trip setting.

The temperature measured by thermocouples on the discharge side of the safety/relief valves must return to within 10 degree F of the temperature recorded before the valve was opened.

The positive change in vessel dome pressure and simulated heat flux occurring within the first 30 seconds after the closure of all MSIV valves must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use beginning of life nuclear data.

If water level reaches the reactor vessel low water level (Level 2) setpoint, RCIC and HPCI shall automatically initiate and reach rated system flow.

Recirculation pump trip shall be initiated if water Level 2 is reached.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.23 2STP-26, RELIEF VALVES

OBJECTIVES

The objectives of this test are a) to verify that the Relief Valves function properly (can be manually opened and closed), b) to verify that the Relief Valves reseat properly after actuation, c) to verify that there are no major blockages in the Relief Valve discharge piping, a d) to demonstrate system stability to Relief Valve operation.

ACCEPTANCE CRITERIA

Level 1

There should be a positive indication of steam discharge during the manual actuation of each Relief Valve.

The flow through each Relief Valve shall compare favorably with value assumed in the FSAR accident analysis at normal operating Reactor pressure.

Level 2

Pressure contro' cem-related variables may contain oscillatory mo response. In these cases, the decay ratio of each uncontrolled mode of response must be less than or equal to 0.25.

The temperature measured by the thermocouples on the discharge side of the valves shall return to within 10 DEG F of the temperature recorded before the valve was opened.

During the rated pressure functional test, the steam flow through each Relief Valve, as measured by Generator Gross MWe, shall not be lower than the average valve response by more than 0.5% of rated MWe.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.24 2STP-27, MAIN TURBINE TRIP

OBJECTIVES

The objectives of this test are to demonstrate the response of the Reactor and its control systems to protective trips of the Main Turbine and to evaluate the response of the bypass and safety/relief valves.

ACCEPTANCE CRITERIA

Level 1

For Turbine and Generator Trips at power levels greater than 50% Nuclear Boiler Rated, there should be a delay of less than 0.1 seconds following the beginning of 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% 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 lines following these transients.

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 Rated Value.

The recirculation pump and motor time constants of the two-pump drive flow coastdown transient should be $\leq$2.5 seconds from 1/4 to 2 seconds after the pumps are tripped.

The total time delay from the start of the Turbine Stop Valve or Control Valve motion to the complete suppression of the electrical arc between the fully open contacts of the RPT circuit breakers shall be less than or equal to 175 milliseconds.

Level 2

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 closure.

The positive change in vessel dome pressure occurring within the first 30 seconds after the initiation of either Generator or Turbine Trip must not exceed predicted values.

The positive change in simulated Heat Flux occurring within the first 30 seconds after the initiation of either Generator or Turbine Trip must not exceed predicted values.

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

Low (L2) water level recirculation pump trip, HPCI and RCIC shall not be initiated.

The temperature measured by thermocouples on the discharge side of the Relief Valves must return to within 10 Degree F of the temperature recorded before the valve was opened.

For the Turbine Trip within the Bypass Valves capacity, the Reactor shall not scram.

The measured Bypass Valve capability shall be equal to or greater than that used in the FSAR analysis (25% of Nuclear Boiler Rated Steam Flow).

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.25 2STP-28, SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

OBJECTIVES

The objectives of this test are to demonstrate that the Reactor a) can be safely shutdown from outside the Control Room, b) can be maintained in a Hot Standby condition from outside the Control Room and c) can be safely cooled from hot to cold shutdown from outside the Control Room. In addition, it will provide an opportunity to demonstrate that the procedures of Remote Shutdown are clear and comprehensive and that operational personnel are familiar with their applications.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

During a simulated Control Room evacuation, the Reactor must be brought to the point where cooldown is initiated and under control, and Reactor vessel pressure and water level are controlled using equipment and controls located outside the Control Room.

The Reactor can be safely shutdown to a Hot Standby condition from outside the Control Room using the minimum shift crew complement.

The Reactor coolant temperature and pressure can be lowered sufficiently (at a rate that does not exceed the Technical Specification Limit) from outside the Control Room to permit operation of the Shutdown Cooling Mode of the Residual Heat Removal System.

The Shutdown Cooling Mode of the Residual Heat Removal System can be initiated from outside the Control Room with a heat transfer path established to the Ultimate Heat Sink.

The Shutdown Cooling Mode of the Residual Heat Removal System can be used to reduce Reactor coolant temperature at a rate which does not exceed the Technical Specification Limit.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.26 STP-29, RECIRCULATION FLOW CONTROL SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate the flow control capability of the plant over the entire pump speed range, in both Individual Local Manual and Combined Master Manual operation modes and to determine that the controllers are set of the desired system performance and stability.

ACCEPTANCE CRITERIA

Level 1

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

Level 2

A scram shall not occur due to Recirculation flow control maneuvers. The APRM neutron flux trip avoidance margin shall be $\geq$7.5% when the power maneuver effects are extrapolated to those that would occur along the 100% rated rod line.

The decay ratio of any oscillatory controlled variable must be $\leq$0.25.

Steady-state limit cycles (if any) shall not produce turbine steam flow variations greater than $\pm$0.5% of rated steam flow.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.27 2STP-30, RECIRCULATION SYSTEM

OBJECTIVES

The objectives of this test are to:

Obtain recirculation system performance data during steady-state conditions, pump trip, flow coastdown, and pump restart.

Verify that the feedwater control system can satisfactorily control water level on a single recirculation pump trip without a resulting turbine trip and associated scram.

Record and verify acceptable performance of the circuit of a two-recirculation pump trip.

ACCEPTANCE CRITERIA

Level 1

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

The recirculation pump and motor time constant of the two pump drive flow coastdown transient should be ≤2.5 seconds from 1/4 to 2 seconds after the pumps are tripped and ≥3.0 seconds from 1/4 to 3 seconds after the pumps are tripped.

Level 2

The reactor water level margin to avoid a high level trip shall be ≥3.0 inches during the one pump trip.

The APRM margin to avoid a scram shall be ≥7.5% during the pump trip recovery.

The core flow shortfall shall not exceed 5% at rated power.

The measured core delta P shall not be >0.6 PSI above prediction.

The drive flow shortfall shall not exceed 5% at rated power.

The measured recirculation pump efficiency shall not be >8% points below the vendor tested efficiency.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.28 2STP-31, LOSS OF TURBINE GENERATOR AND OFFSITE POWER

OBJECTIVES

This test determines electrical equipment and reactor system transient performance during a loss of main turbine-generator coincident with loss of all sources of offsite power.

ACCEPTANCE CRITERIA

Level 1

All safety systems, such as the Reactor Protection system, the diesel-generators, and HPCI must function properly without manual assistance, and HPCI and/or RCIC system action, if necessary, shall keep the reactor water level above the initiating level of Low Pressure Core Spray, LPCI, Automatic Depressurization System, and MSIV Closure. Diesel generators shall start automatically.

Level 2

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

Reactor pressure shall not exceed 1250 psig.

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

Normal cooling systems shall be capable of maintaining adequate drywell cooling and adequate suppression pool water temperature.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.29 2STP-32, ESSENTIAL HVAC SYSTEM OPERATION AND CONTAINMENT HOT PENETRATION TEMPERATURE VERIFICATION

OBJECTIVES

The objectives of this test are to demonstrate, under actual/normal operating conditions, that the various HVAC systems will be capable of maintaining specified ambient temperatures and relative humidity within the following areas:

a) Primary Containment (drywell and suppression chamber)

b) Reactor Enclosure and Main Steam Tunnel

c) Control Room

d) Control Enclosure

e) Radwaste Enclosure

This test also verifies that the concrete temperature surrounding containment hot penetrations remains within specified limits.

In addition, this test shall verify that the concrete temperature surrounding Main Steam and Feedwater containment penetrations remains within specified limits.

ACCEPTANCE CRITERIA

Level 1

The drywell area volumetric average air temperature is not to exceed 135 degrees F.

Level 2

The drywell area and suppression chamber are maintained between 65 degrees F and 150 degrees F.

The reactor pressure vessel (RPV) support skirt surrounding air temperature is maintained above a minimum of 70 degrees F.

The concrete temperatures surrounding primary containment Main Steam line and Feedwater line penetrations are maintained at less than or equal to 200 degrees F.

The following areas of the control enclosure are maintained between 65 degrees F and 104 degrees F: rooms 164, 258, 263, 336, 428, 429, 430, 431, 432, 433, 434, 435, 449, 450, 452, 453, 454, 540, 619, 625, 614A and 624B.

The control room is maintained at a temperature between 65 degrees F and 78 degrees F and relative humidity between 30% R.H. and 90% R.H.

The following areas of the reactor enclosure are maintained between 65 degrees F and 104 degrees F: rooms 182, 18(, 279, 284, 287, 370, 475, 479, 574, 580B, 580C, 580D, 580G, 581, 582, 583, 585, 594, 637, 638, 641, 651 and 653.

The following areas of the reactor enclosure are maintained between 65 degrees F and 110 degrees F: rooms 576, 577, 578 and 579.

The following areas of the reactor enclosure are maintained between 65 degrees F and 115 degrees F: rooms 173, 174, 179, 180, 181, 184, 185, 188, 280, 281, 283, 285, 575, 584, 589, 593, and 597.

The following areas of the reactor enclosure are maintained between 65 degrees F and 120 degrees F: rooms 286, 374, 375, 375, 480 and 587.

The following areas of the Radwaste Enclosure are maintained between 65 degrees F and 76 degrees F: rooms 410, 411, 412, 415, 416, 417 and 418.

The following battery control rooms are to be maintained between 65 degrees F and 104 degrees F: rooms 323, 324, 360, 361, 425, 426, 427 and 436.

The auxiliary equipment room 542 is to be maintained between 60 degrees F and 82 degrees F, with the relative humidity maintained between 30% R.H. and 90% R.H.

The reactor pressure vessel support skirt flow impingement velocity is less than 15 per second.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.30 2STP-33, PIPING STEADY STATE VIBRATION

OBJECTIVE

The objective of this test is to verify that the steady state vibration of Main Steam, Reactor Recirculation and selected BOP piping systems is within acceptable limits.

ACCEPTANCE CRITERIA

Level 1

Operating Vibration: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the allowable value of that point.

Level 2

Operating Vibration: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the expected value of that point.

The steady state vibrations of visually examined balance of plant piping are acceptable if the vibration levels are judged by a qualified test engineer to be negligible. Vibration levels judged to be potentially significant are evaluated as determined necessary by BPC Project Engineering.

The vibration measured by a remote accelerometer is acceptable if the acceleration frequency spectrum falls in the negligible region of the acceptance chart of that accelerometer. If the acceleration frequency spectrum crosses the negligible region boundary, the test results shall be evaluated by BPC Project Engineering.

RESULTS

The new fuel design did not affect the performance of systems to satisfy the acceptance criteria of this test.

2.31 2STP-34, OFFGAS PERFORMANCE VERIFICATION

OBJECTIVES

The objectives of this test are to verify that the Offgas Recombination and Ambient Charcoal System operates within the technical specification limits and expected operating conditions.

ACCEPTANCE CRITERIA

Level 1

The allowable dose and dose rates from releases of radioactive gaseous and particulate effluents to areas at and beyond the SITE BOUNDARY shall not be exceeded.

Allowable limits on the radioactivity release rates of the six noble gases measured at the after condenser discharge shall not be exceeded.

The hydrogen content of the offgas effluent downsteam of the recombiner shall be equal to or less than 4% by volume.

The total flow rate of dilution steam plus offgas when the steam jet air ejectors are in operation shall exceed 9555 lbs/hr.

Level 2

System flows, pressures, temperatures and dewpoint shall be within expected performance values.

The preheater, catalytic recombiner, after condenser, Hydrogen Analyzers, cooler condenser, activated charcoal beds and the HEPA filter shall be performing their required functions adequately. The automatic drain systems function adequately.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.32 2STP-35, RECIRCULATION SYSTEM FLOW CALIBRATION

OBJECTIVES

The objectives of this test are to perform a complete calibration of the recirculation system flow instrumentation, including specific signals to the plant process computer and to adjust the recirculation flow control system to limit maximum core flow to 107% of rated core flow.

ACCEPTANCE CRITERIA

Level 1

None

Level 2

Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide 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 107% of rated.

The calculated jet pump M-Ration shall not be < 0.2 points below prediction.

The nozzle and riser plugging criteria shall not be exceeded.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.33 2STP-36, PIPING DYNAMIC TRANSIENTS

OBJECTIVES

The objectives of this test are to verify that the following pipe systems are adequately designed and restrained to withstand the following respective transient loading conditions:

Main Steam - Main Turbine Stop Valve/Control Valve closures at 20-25%, and 95-100% of rated thermal power.

Main Steam and Relief Valve Discharge - Main Steam Relief Valve actuation.

Recirculation - Recirculation Pump trips and restarts.

High Pressure Coolant Injection steam supply - High Pressure Coolant Injection turbine trips.

Feedwater - Reactor feed pump trips/coastdowns.

ACCEPTANCE CRITERIA

Level 1

Operating Transients: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the allowable value of that point.

Level 2

Operating Transients: The measured amplitude (peak to peak) of each remotely monitored point shall not exceed the expected value of that point.

The maximum measured loads, displacements, and/or velocities are less than or equal to the acceptance limits specified.

In the judgment of the qualified test engineers, no signs of excessive piping response (such as damaged insulation; markings on piping, structural or hanger steel, or walls; damaged pipe supports; etc.) are found during a post-transient walkdown and visual inspection of the piping tested and associated branch lines.

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.34 2STP-70, REACTOR WATER CLEANUP SYSTEM

<u>OBJECTIVES</u>

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

<u>ACCEPTANCE CRITERIA</u>

Level 1

None

Level 2

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

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

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

<u>RESULTS</u>

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.

2.35 2STP-71, RESIDUAL HEAT REMOVAL SYSTEM

OBJECTIVES

The objectives of this test are to demonstrate the ability of the Residual Heat Removal (RHR) System to remove residual and decay heat from the nuclear system so that refueling and nuclear servicing can be performed. Additionally, this test will demonstrate the ability of the RHR System to remove heat from the suppression pool.

Level 1

The RHR System shall be capable of operating in the Suppression Pool Cooling Mode at the heat exchanger capacity specified.

Level 2

None

RESULTS

The new fuel design did not affect the performance of systems needed to satisfy the acceptance criteria of this test.