THE FORT ST. VRAIN

INITIAL APPROACH TO

POWER TESTS (B-SERIES)

INTERIM REPORT 20

Report for Period Ending August 22, 1981

TABLE OF CONTENTS

INITIAL APPROACH TO POWER TESTS

(B SERIES STARTUP TESTS)

The initial approach to power is accomplished in a series of discrete power level stages. At each power level, tests are made to measure the characteristics of the plant and to ensure that the plant is within its design limits, and the power can be safely increased to the next stage.

The initial phase of the approach-to-power program will increase the reactor power and steam conditions in stages until approximately 28% power when rated steam conditions are achieved. From this level to full power, the reactor power is increased in stages maintaining rated steam conditions. The sequence for the performance of these tests is given in Figure 1, together with the corresponding approximate reactor power levels. The reactor power levels, helium flow rates, feedwater flow rates, steam temperatures, and steam pressure given in the following description of the initial approach-to-power may differ somewhat from those in the actual approach to power due to change in test requirements or improvements in operating methods identified during other tests.

In general, the initial approach-to-power will be accomplished in the following order:

1. Feedwater flow will first be established through both steam generator loops and the bypass flash tank system using a boiler

feedpump. Helium flow through the core will be provided using one circulator in each loop.

2. The reactor power will be increased to approximately 2%.

3. The reactor power, feedwater flow, and helium flow rate will be simultaneously increased to 5% power, 20% helium flow, and 25% feedwater flow using reactor generated steam from the bypass flash tank supplemented by the auxiliary boiler to power the circulating turbines, turbine driven boiler feedpump, and other plant steam requirements.

4. The reactor power will then be increased to approximately 8%, concurrent with an increase in feedwater flow to about 30%. The helium flow will be maintained at about 20% during this power increase. At this condition, the second circulator in each loop will be started maintaining constant helium flow, and the main steam pressure will be increased to 2,400 psig.

5. The reactor power will be increased to about 11%, and feedwater will be reduced to 25% to initiate boiling.

6. The reactor power will be increased to about 18% simultaneously with an increase in helium flow to about 33%, maintaining 25% feedwater flow, followed by an increase in reactor power to about 26% with a helium flow of 49%. At this condition, the main steam temperature will be about 800 degrees fahrenheit.

7. The helium flow will then be reduced to about 40% concurrent with a slight adjustment of the reactor power to about 28%.

8. The reactor power will be increased in stages to about 40%, 50%, 60%, 80%, and finally to 100% of full power. During these power level increases, the helium flow rate through the core will be increased to maintain full steam conditions.

This report covers tests performed between 70% and approximately 88% reactor power.

Each power level was maintained for a period of time to perform one or more of the following tests. Preliminary analysis of these measurements, as specified in the overall controlling test document, was completed prior to increasing the reactor power to the next stage.

Steam System Performance Tests (B-1)

Just prior to steaming, and at subsequent power levels during the initial rise-to-power, data will be accumulated and analyzed on the performance of the steam generators, the turbine, and the steam plant auxiliaries. Measurements of the turbine performance will be made at the lower power levels, and the turbine will be loaded at about 28% reactor power.

Analysis of Chemical Impurities in the Pr ary Coolant (B-2)

As the reactor power level is increased to about 11% of rated, the core and reactor internals will experience temperatures in excess of those reached during the core heat-up for reactivity coefficient measurements. At these temperatures, additional impurities will be degassed. Data on the performance of the helium purification system in removing these chemical impurities from the primary coolant will be taken and analyzed.

PCRV Performance Tests (B-3)

As the reactor power level is increased to 28% power, the helium pressure and temperatures approach their quarter load values which results in a system heat load of approximately 80%. At each power level stage up to 28% power, and at selected stages up to full power, data will be taken and analyzed on the performance of the PCRV and its cooling system on the structural response of the PCRV to increased internal pressure and on the primary system helium use rate.

Primary Coolant System Performance Tests (B-4)

At each power level, data on the performance of the helium circulators and their auxiliaries will be taken and analyzed. Measurements of the radial power distribution (region peaking factors) will be made at approximately 2%, 5%, and 8% reactor power. Data on the performance and calibration of the core helium flow orifice valve will be obtained at approximately 28%, 50%, and 100% reactor power.

Plant Instrumentation Performance Tests (B-5)

In these tests, the performance of the portions of the plant instrumentation, which could not be tested prior to power operation, will be checked. The nuclear instrumentation will be calibrated by means of heat balance measurements and analyses. The calibration of the condensate and feedwater flow instrumentation and the core region outlet thermocouples will be checked. The core region outlet thermocouple test will be performed just prior to the first adjustment of the helium flow orifices at approximately 8% power and again at approximately 100% power.

Plant Transient Performance Tests (B-6)

In these tests, the transient performance of the plant will be tested and analyzed. The testing will include: a scram and turbine trip from approximately 28% reactor power with rated steam conditions, a turbine trip from approximately 40% reactor power, a main turbine generator load rejection from approximately 60% reactor power to house load, sequential tripping of the two circulators in a loop from

approximately 80% reactor power and resultant loop shutdown, and boiler feedpump start and stop transients.

Plant Automatic Control System Performance Tests (B-7)

The components of the automatic control system will be placed into service and tested as the controlled variables come into their controllable range. Dynamic verification tests of the control system will be performed at selected power levels during the power level increase of the initial approach to full power. A demonstration of full load change from approximately 100% to approximately 25% turbine load will be made under full automatic control.

Reactor Coefficient Measurements (B-8)

Measurements of changes in reactivity will be made during the approach to full power by measuring the change in control rod positions required to produce a core temperature and reactor power level change.

Differential Control Rod Worth Measurements (B-9)

The reactivity worth of control rods which are moved during the initial rise-to-power will be measured using a reactivity computer to obtain the instantaneous reactivity change produced by a control rod motion.

Xenon Buildup and Decay Measurements (B-10)

The reactivity change produced by buildup, burnout, or decay of xenon poison following a power level change will be measured by recording the change in the critical control rod positions following a change.

Xenon Stability Test (B-11)

In this test, the absence of any sustained xenon oscillations is demonstrated. At 100% power, a perturbation is produced from equilibrium xenon by inserting a control rod in one region and withdrawing a control rod in another region. The indicated power level and region outlet temperatures are recorded as a function of time and analyzed for the presence of any oscillation produced by xenon.

Shielding Surveys (B-12)

At approximately 28% reactor power and approximately 100% reactor power, surveys of the radiation levels within the plant are performed. An additional survey is taken during and following any regeneration of the helium purification system. These measured data are recorded and analyzed to demonstrate the adequacy of the shielding design.

Radiochemical Analysis of the Primary Coolant (B-13)

In this test, the radioactive gaseous fission products in the primary coolant will be sampled and analyzed. These tests are used in the initial startup phase to define fuel fission product release-to-birth ratio at zero burnup and will yield information on the fraction of failed fuel particle coatings. This test is performed at each major power level of the initial rise-to-power.

RISE-TO-POWER TESTING SEQUENCE

REACTOR POWER (% OF RATED)	2%	5%	8%	11%	18%	26%	28%	40%	50%	60%	80%	100%
Steam System Performance			o	o	o	o	o	o	o	o	o	o
Analysis of Chemical Impurities	o	o	o	o	o	o	o	o	o	o	o	o
PCRV Performance	o	o	o	o		o	o	o	o	o	o	o
Primary System Performance	o	o	o	o	o	o	o	o	o	o	o	o
Plant Instrumentation	o	o	o	o	o	o	o	o	o	o	o	o
Plant Transient Performance							o		o	o	o	
Auto Control System	o	o	o	o		o	o	o	o	o	o	o
Reactivity Coefficients	o	o	o	o	o	o	o	o	o	o	o	o
Differential Rod Worths	o	o	o	o	o	o	o	o	o	o	o	o
Xenon Buildup and Decay			o			o			o			o
Xenon Stability												o
Shielding Surveys							o					o
Radiochemical Analysis	o	o	o	o	o	o	o	o	o	o	o	o

FIGURE 1

ACKNOWLEDGEMENT

The contents of this report on the results of B Series Startup Testing at Fort St. Vrain, Unit No. 1, have been taken from unpublished, internal reports of General Atomic Company and Public Service Company of Colorado.

This is an interim report based on preliminary data; and therefore, both data and results are subject to change. This report will be supplemented periodically as further testing is completed.

HISTORICAL SUMMARY OF PLANT OPERATION

The last testing reported was covered in interim report number 8 for the period ending August 22, 1978. After this, the plant continued operation with the Nuclear Regulatory Commission limit of 70% power.

In February, 1979, the shutdown for the first refueling was started with the shutdown extending until June, 1979. After startup, normal operation was resumed until September 1, 1979, when the plant was shutdown due to inconsistencies found in the safety related piping and hangers. This was found during an audit and analysis committed to by the Company and was reported in RO 50-267/79-35/01-T-0.

On September 15, 1979, the reactor was started up and operation continued until October 26, 1979, when a maintenance shutdown began to install the region constraint devices (RCD). The RCD installation was completed, and the reactor taken critical on December 25, 1979. The RCD's are metal clamps that join the tops of all the regions together to maintain nearly uniform gap flow areas between regions.

Following a reactor scram, 1B helium circulator static seal failed in January, 1980, requiring the circulator to be replaced with the spare.

On February 17, 1980, the reactor was taken critical again, and the turbine generator was put on line March 5, 1980.

Operation continued throughout the yea ınd into 1981.

On March 16, 1981, Amendment 23 was issued, which authorized f uctuation testing at greater than 70% power. Functuation testing was started on March 18, 1981, with power levels of 70% to 88% being reached during the period April 17 to April 24, and on May 13, 1981.

On May 13, 1981, the turbine generator tripped on high vibration, due to a loose shroud on the low pressure turbine blading and was followed by a reactor scram. As the turbine repair was to take several weeks, the second refueling outage was started on May 20, 1981. During this period, 1B circulator was replaced again due to another failed static seal.

The refueling shutdown was finished in July, 1981, and the reactor was taken critical on July 13, 1981. Plant operation has continued throughout the rest of this report period at power levels up to 70% and 230 MWe.

TESTING SUMMARY

This report covers the period from May 23, 1981, through August 22, 1981, and the analysis of the test results collected during the last period of February 23, 1981, to May 22, 1981.

Due to the installation of the region constraint devices and Nuclear Regulatory Commission requirements, the B Series Startup Tests, RT-500K (fluctuation testing at greater than 70% power) and RT-485 (control rod drive temperature data collection) were run concurrently, with RT-500K being the controlling document for testing under normal conditions from 40% to 100% power. The test schedule is called out in T-164 (coordination procedure for testing at greater than 70% power) and is included at the end of the test summary section.

Testing at greater than 70% power was initiated on three occasions (April 17, 1981, April 24, 1981, and May 13, 1981) up to a maximum power level of 88%, at which time turbine vibrations necessitated a plant shutdown and the second refueling outage was begun.

Based on tests completed so far, the following conclusions can be drawn concerning plant behavior (for details refer to individual B Series Test Reports).

1. The steam generator performance at 80% reactor power was acceptable. The maximum measured helium hot streak temperature was less than or equal to 30 degrees fahrenheit about the mixed mean inlet helium temperature, and the maximum main steam subheader deviation within a steam generator module was within the 18 degrees fahrenheit allowable deviation. The maximum measured crossover tube temperature was 849 degrees fahrenheit, which was well below the allowable 950 degrees fahrenheit limit. The average reheat steam outlet temperature was 991 degrees fahrenheit, the temperature controller was set for 990 degrees fahrenheit. The average main steam outlet temperature was 997 degrees fahrenheit. The performance of the main turbine generator at 80% reactor power was acceptable. (SUT B-1)

2. During the performance of this test, variations of the primary system impurity levels were experienced due to problems with continued off-gassing and moisture ingress resulting from circulator trips. Although the helium purification system was in operation, total oxidants were in excess of 10 ppm during the testing periods. See Reportable Occurrence Report No. 50-267/81-027. (SUT B-2)

3. Based upon the PCRV heat load data taken during the test, the maximum heat load relative to the acceptance criteria was experienced in the top penetration area of the PCRV. The heat load in this area was 104% of the acceptance criteria value. The

heat loads in all other areas of the prestressed concrete reactor vessel were lower than design values. This situation is similar to that observed in all previous test data for B-3, Section 2R. (SUT B-3)

4. The maximum PCRV load cell change relative to other corresponding load cells was the top head cross head tendon TIR-M1. However, it is still well above the recommended low limit. It is recommended that load cell TIR-M1 be included in the next scheduled lift off test. The zero psig reference data was obtained nearly two years ago and is most likely no longer valid due to load cell changes and long term PCRV creep. ('UT B-3)

5. Primary coolant makeup rate is higher than expected due to a known leak in the Loop 2 steam generator interspace. Purified helium leaks into the reheat steam piping. This was explained and reported in RO 50-267/80-30/03-L-0. (SUT B-3)

6. The measured values of: circulator pr::ssure rise, circu...or helium flow, and core pressure drop continue to deviate from the anticipated values. The measured circula:or pressure rise was 40% below predicted, consistent with the pre/ious trends. The helium flow was 9% higher than predicted to match main and reheat steam temperatures, and the core pressure drop was 46% below predicted. After installation of the region constraint devices,

fluctuation testing at greater than 70% reactor power was initiated.

Testing performed to date indicates that the region constraint devices do prevent the onset of fluctuations. However, a single core redistribution has been noticed each time power is increased towards 100%.

Future testing will be done as plant conditions permit. (SUT B-4)

7. The expected decalibration of the nuclear instrumentation as a function of control rod pattern continues to be observed. The nuclear channels are checked daily and at every 10% power level change, and adjusted as required to agree with a heat balance. (SUT B-5)

8. The shielding survey at 80% power showed less than 1 mr/hour in all locations except the south wall of the hot service facility, which read 2.8 mr/hour, and the gas waste vacuum tank which read 1.5 mr/hour. All the locations were well within anticipated. (SUT B-12)

9. The primary coolant radioactivity level and the fission product release to birth ratio (R/B) continue to be less than

anticipated. At 80% power, Ba-139 was observed in the fast gas sample. (SUT B-13)

STEADY STATE B-O DATA COLLECTION*

		APPROXIMATE POWER LEVELS				
SUT	TITLE	~60	~70	~80	~90	~100
B1-.	Steam Generator Steady State	O	O	O	O	R
B1 3	Steam Generator Steady State (Data with Plugged Tube)			R		
B1-4	Steam Generator Steady State (HP Feedwater Heater Bypassed)			R		
B1-6	Turbine Generator Steady State Performance	O	O	O	O	R
B2-1	Primary Coolant System Impurities	O	O	R	O	R
B2-2	Purification Chemical Impurities	O	O	R	O	R
B2-3	Calibration of Gas Chromatograph	O	O	R	O	R
B3-1R	Liner Cooling Maximum Temperatures	O	O	R	O	R
B3-1S	Liner Cooling ADJ (As Required)	-	-	-	-	-
B3-1Q	PCRV Liner Cooling	O	O	O	O	R
B3-2Q	PCRV Data Scan	O	O	O	O	R
B3-2R	PCRV Internal Temperatures	O	O	R	O	R
B3-3A	PCRV Leak Tightness	O	O	R	O	R
B3-3B	Full Pressure PCRV Leakage SR 5.2.16a-M (as required)	-	-	-	-	R
B4-1(O)	Circulator Primary Coolant	O	O	O	O	R
B5-1	Nuclear Instrument Calibration	O	O	O	O	R
B5-2	Core Region TC Calibration	-	-	-	-	R
B5-3	Feedwater Flow Calibration	O	O	O	O	R
B11	Xenon Stability Test					R
B12-2	Shielding Survey	O	O	O	O	R
B13-1	Radioactive Gas Analysis	O	O	O	O	R
B13-2	Iodine Probe Analysis	-	-	-	-	R
*RT-485	Control Rod Drive Internal Temperatures *(Data to be taken at 6% Intervals Between 70 and 100%)	O	-	-	-	-
RP390-M	Generator Temperature Monitoring	O	O	O	O	O

"O" Optional Data Taking
"R" Required Data Taking

*NOTE: For further details on individual B-O steady state data taking frequency see pages 10-15.

FIGURE 1

STEAM SYSTEM PERFORMANCE VERIFICATION (B-1)

Data

1. Typical steam generator temperature data taken at approximately 80% is shown in Table B-1.1.

2. Typical main turbine parameters at about 80% reactor thermal power (80% turbine load) are shown in Table B-1.2.

TABLE B-1.1

STEAM GENERATOR MODULE TEMPERATURES

Module	Inlet Helium Temperature	Cold Reheat Steam Temperature °F	Reheat Steam Temperature	Main Steam Temperature
B-1-1	1347	546	998	980 (2)(3)
B-1-2	1319	539	980	997 (1)
B-1-3	1346	543	1001	1002
B-1-4	1330	540	990	1002
B-1-5	1340	544	994	1000
B-1-6	1333	541	992	995
B-2-1	1347	526	990	996
B-2-2	1332	527	998	1000
B-2-3	1342	523	991	990 (3)
B-2-4	1357	530	1005	1000 (2)
B-2-5	1340	526	978	1000 (4)
B-2-6	1323	525	985	998 (1)

(1) Low helium temperature to B-1-2 and B-2-6.

(2) High helium temperature to B-2-4 and B-1-1.

(3) Trim on B-1-1 and B-2-3 trim valve needs to be corrected.

(4) Possible high attemperation flow to B-2-5.
Low hot reheat steam at module outlet.

INSTRUMENT	DATA
Linear Reactor Power Level	79%
Wide Range Reactor Power Level	100%
Helium Pressure	665 PSIA
Helium Pressure	660 PSIA
Helium Pressure	680 PSIA
Circulator Inlet Temperature	688°F
Circulator Speed, 1A	7500 RPM
Circulator Speed, 1B	7600 RPM
Circulator Speed, 1C	7600 RPM
Circulator Speed, 1D	7900 RPM
Circulator Helium Flow, 1A	74%
Circulator Helium Flow, 1B	76%
Circulator Helium Flow, 1C	82%
Circulator Helium Flow, 1D	75%
Circulator Helium Differential Pressure, 1A	4.5 PSI
Circulator Helium Differential Pressure, 1B	4.4 PSI
Circulator Helium Differential Pressure, 1C	4.4 PSI
Circulator Helium Differential Pressure, 1D	4.5 PSI
Feedwater Temperature, Loop 1	370°F
Feedwater Temperature, Loop 2	370°F
Feedwater Flow, Loop 1	860,000 lbs/hr
Feedwater Flow, Loop 2	860,000 lbs/hr
Cold Reheat Pressure Before Circulators, Loop 1	695 PSIG
Cold Reheat Pressure Before Circulators, Loop 2	695 PSIG
Cold Reheat Pressure After Circulators, Loop 1	460 PSIG

INSTRUMENT	DATA
Cold Reheat Pressure After Circulators, Loop 2	450 PSIG
Main Steam Pressure, Loop 1	2500 PSIG
Main Steam Pressure, Loop 2	2450 PSIG
Main Steam Temperature, Loop 1	1005°F
Main Steam Temperature, Loop 2	1010°F
Turbine First Stage Pressure	1200 PSIG
Turbine Throttle Pressure	2450 PSIG
Hot Reheat Steam Pressure, Loop 1	440 PSIG
Hot Reheat Steam Pressure, Loop 2	430 PSIG
Desuperheater Water Flow, Loop 1	45,000 lbs/hr
Desuperheater Water Flow, Loop 2	45,000 lbs/hr
Feedwater Valve Differential Pressure, Loop 1	120 PSID
Feedwater Valve Differential Pressure, Loop 2	100 PSID
Hot Reheat Pressure	415 PSIG
Hot Reheat Temperature at Turbine Generator Inlet	986°F
Steam Chest Pressure	2430 PSIG
Flash Tank Pressure	700 PSIG
Turbine Generator Load	270 MW
Module Feedwater Flow, Channel 1	67.0%
Module Feedwater Flow, Channel 2	63.5%
Module Feedwater Flow, Channel 3	66.5%
Module Feedwater Flow, Channel 4	65.5%
Module Feedwater Flow, Channel 5	66.0%
Module Feedwater Flow, Channel 6	66.0%
Module Feedwater Flow, Channel 7	72.5%

INSTRUMENT	DATA
Module Feedwater Flow, Channel 8	66.0%
Module Feedwater Flow, Channel 9	69.0%
Module Feedwater Flow, Channel 10	71.5%
Module Feedwater Flow, Channel 11	68.5%
Module Feedwater Flow, Channel 12	69.5%
Module Main Steam Temperature, Channel 1	980°F
Module Main Steam Temperature, Channel 2	997°F
Module Main Steam Temperature, Channel 3	1002°F
Module Main Steam Temperature, Channel 4	1002°F
Module Main Steam Temperature, Channel 5	1000°F
Module Main Steam Temperature, Channel 6	995°F
Module Main Steam Temperature, Channel 7	996°F
Module Main Steam Temperature, Channel 8	1000°F
Module Main Steam Temperature, Channel 9	999°F
Module Main Steam Temperature, Channel 10	1000°F
Module Main Steam Temperature, Channel 11	1000°F
Module Main Steam Temperature, Channel 12	998°F
Circulator Outlet Steam Temperature, Loop 1	650°F
Circulator Outlet Steam Temperature, Loop 2	632°F
Reheater Inlet Temperature, Channel 1	546°F
Reheater Inlet Temperature, Channel 2	539°F
Reheater Inlet Temperature, Channel 3	543°F
Reheater Inlet Temperature, Channel 4	540°F
Reheater Inlet Temperature, Channel 5	544°F
Reheater Inlet Temperature, Channel 6	541°F

INSTRUMENT	DATA
Reheater Inlet Temperature, Channel 7	526°F
Reheater Inlet Temperature, Channel 8	527°F
Reheater Inlet Temperature, Channel 9	523°F
Reheater Inlet Temperature, Channel 10	530°F
Reheater Inlet Temperature, Channel 11	526°F
Reheater Inlet Temperature, Channel 12	525°F
Reheater Outlet Temperature, Channel 1	998°F
Reheater Outlet Temperature, Channel 2	980°F
Reheater Outlet Temperature, Channel 3	1001°F
Reheater Outlet Temperature, Channel 4	990°F
Reheater Outlet Temperature, Channel 5	994°F
Reheater Outlet Temperature, Channel 6	992°F
Reheater Outlet Temperature, Channel 7	990°F
Reheater Outlet Temperature, Channel 8	988°F
Reheater Outlet Temperature, Channel 9	991°F
Reheater Outlet Temperature, Channel 10	1005°F
Reheater Outlet Temperature, Channel 11	978°F
Reheater Outlet Temperature, Channel 12	985°F

TABLE B-1.2

MAIN TURBINE GENERATOR OPERATING PARAMETERS

Item	Predicted Values	Measured Data
Load MW(e)	268	264
Main Steam Temperature	1,000	990
Reheat Steam Temperature	1,000	1000
First Stage Pressure, PSIG	1,249	1230
Main Steam Pressure	2,400	2390
Cold Reheat Pressure	675	670
Feedwater Temperature	386	375
Condenser Pressure	2.5	2.25
Turbine Speed	3,600	3,650
Attemperation Flow	110,200	90,000
Maximum Vibration (Pt. 5)	---------	3.65 Mils

INSTRUMENT	DATA
Load	264 MW
Number of Hours at Load	2 Hours
Reactive KVA	10 Mega Vars
Hydraulic Piston Stroke (Control Valves)	100%
Hydraulic Piston Stroke (Control Valves)	100%
Hydraulic Piston Stroke (Control Valves)	13%
Hydraulic Piston Stroke (Control Valves)	0%
Steam Chest Pressure	2390
Main Steam Temperature	990°F
First Stage Pressure	1230 PSIG
High Pressure Exhaust to Reheater	670 PSIG
High Pressure Exhaust to Reheater	710°F
Reheat Bowl Pressure	380°F
Hot Reheat Temperature	1000°F
Hot Reheat Steam to IP Turbine Pressure	420 PSIG
Cold Reheat Steam PT #14	690°F
Fifteenth Stage Pressure, Heater 1	6.9
Fourteenth Stage Pressure, Heater 2	11.0
Thirteenth Stage Pressure, Heater 3	20.5
Eleventh Stage Pressure, Heater 4	64
Tenth Stage Pressure, Heater 5	116
Eighth Stage Pressure, Heater 6	185
Feedwater Temperature Leaving Top Heater	375
Exhaust Pressure	23 Hg
Barometer	25.25 Hg

INSTRUMENT	DATA
Steam Flow	1.6×10^6 lbs/hr
Condensate Flow	1.64×10^6 lbs/hr
Quantity of Makeup Demineralizer Water	53 GPM
Feedwater Heaters in Service	All But #5
Quantity of Attemperation to Reheat Loop 2	45×10^3 lbs/hr
Quantity of Attemperation to Reheat Loop 1	45×10^3 lbs/hr
Main Steam Pressure	2430
Main Steam Pressure	2450
Operating Oil Pressure	225
Bearing Header Pressure	25
Main Pump Suction Pressure	23
Turbine Speed	3650
Armature Current	6950
Armature Current	7050
Armature Current	6950
Armature Voltage	2190 KV
Field Current	1700 Amps
Field Voltage	200 Volts
Field Temperature	54°C
Hydrogen Pressure	19 PSIG
Hydrogen Purity	98%
Differential Fan Pressure, Water	3.1
Exhaust Hood Temperature	112
Exhaust Hood Temperature	110
Cold Gas Temperature	32°C

INSTRUMENT	DATA
Cold Gas Temperature	34°C
Cold Gas Temperature	31°C
Cold Gas Temperature	32°C
Stator Temperature, Liquid Out	36°C
Stator Temperature, Liquid In	40.5°C
Stator Temperature, Machine Gas Temperature	37°C
Exciter Air In Temperature	39°C
Exciter Air Out Temperature	28.5°C
Collector Air Out Temperature	38.5°C
Turbine Temperature and Differential Temperature Data Point 1	1010°F
Turbine Temperature and Differential Temperature Data Point 2	1010°F
Turbine Temperature and Differential Temperature Data Point 3	1000°F
Turbine Temperature and Differential Temperature Data Point 4	960°F
Turbine Temperature and Differential Temperature Data Point 5	860°F
Turbine Temperature and Differential Temperature Data Point 6	740°F
Turbine Temperature and Differential Temperature Data Point 7	990°F
Turbine Temperature and Differential Temperature Data Point 8	840°F
Turbine Temperature and Differential Temperature Data Point 9	135°F
Turbine Temperature and Differential Temperature Data Point 10	510°F
Turbine Temperature and Differential Temperature Data Point 11	365°F

INSTRUMENT	DATA
Turbine Temperature and Differential Temperature Data Point 12	160°F
Turbine Temperature and Differential Temperature Data Point 13	710°F
Turbine Temperature and Differential Temperature Data Point 14	710°F
Turbine Temperature and Differential Temperature Data Point 15	690°F
Turbine Temperature and Differential Temperature Data Point 16	700°F
Turbine Temperature and Differential Temperature Data Point 17	860°F
Turbine Temperature and Differential Temperature Data Point 18	840°F
Turbine Temperature and Differential Temperature Data Point 19	130°F
Turbine Temperature and Differential Temperature Data Point 20	60°F
Turbine Temperature and Differential Temperature Data Point 21	55°F
Turbine Temperature and Differential Temperature Data Point 22	140°F
Turbine Vibration Data Point 1	0.12 Mils
Turbine Vibration Data Point 2	0.09 Mils
Turbine Vibration Data Point 3	2.70 Mils
Turbine Vibration Data Point 4	0.92 Mils
Turbine Vibration Data Point 5	3.65 Mils
Turbine Vibration Data Point 6	1.93 Mils
Turbine Vibration "A", #1, Data Point 7	1.3 Mils
Turbine Vibration "A", #2, Data Point 8	.35 Mils
Turbine Vibration "C", #1, Data Point 9	.6 Mils
Turbine Vibration "C", #2, Data Point 10	2.4 Mils

CHEMICAL IMPURITIES IN THE PRIMARY COOLANT (B-2)

Data

Table B-2.1 shows the data collected during operation at greater than 70% power.

TABLE B-2.1

DATE	TIME	CORE OUTLET TEMPERATURE °F	REACTOR POWER %	O_2 PPMV	CO PPMV	CO_2 PPMV	H_2 PPMV	CH_4 PPMV	N_2 PPMV	H_2O PPMV
4-16-81	1600	1353	70	0.0	13.2	3.3	5.6	0.4	28.8	0
	2000	1359	70	0.0	11.8	3.1	8.0	0.3	27.4	0
4-17-81	0200	1363	74	2.3	11.6	2.1	4.8	0.2	27.6	0
	0400	1352	70	0.0	11.8	2.1	5.6	0.4	25.9	0
	1000	1371	74	1.5	12.2	1.8	8.0	0.4	24.5	0
	1200	1378	70	1.5	12.2	1.5	8.0	0.2	24.5	0
	1400	1386	80	0.8	11.7	1.4	8.0	0.4	24.5	0
	1600	1362	74	1.1	11.9	1.7	9.6	0.6	24.5	0
	1800	1364	74	0.0	10.7	1.5	7.2	0.4	67.7	0
	2200	1374	80	0.8	13.0	1.7	8.0	0.2	71.0	0
4-18-81	0200	1371	80	0.0	12.5	1.7	8.0	0.2	52.8	0
	0600	1371	80	1.1	11.2	2.1	7.7	0.3	41.0	0
	0800	1371	80	1.1	12.9	1.7	8.0	0.2	52.8	0
	1200	1372	80	0.0	12.9	1.7	6.4	0.4	40.3	0
	2000	1374	80	0.0	10.2	1.4	7.2	0.2	43.2	0
4-24-81	0900	1369	80	1.0	11.6	1.8	11.1	0.2	71.0	0
	1000	1370	80	1.4	13.1	1.1	12.8	0.2	65.3	0
	1200	1380	84	0.7	13.7	1.2	11.1	0.2	73.0	0
	1400	1366	80	0.7	13.8	1.8	11.1	0.1	55.7	0
	1600	1381	84	2.8	13.5	1.0	11.9	0.4	47.0	0
	1800	1392	88	2.8	17.4	1.8	15.3	0.5	57.6	0
	2000	1392	84	3.2	15.9	1.6	17.0	0.4	51.8	0
	2200	1365	84	2.8	13.5	2.6	11.9	0.3	46.1	0
4-25-81	0000	1364	80	0.7	15.3	1.5	12.8	0.2	72.0	0

PCRV PERFORMANCE TESTS (B-3)

Section A - PCRV Liner Cooling System

Data

The heat loads for each subsection of the PCRV at 80% power are shown in Table B-3A.1 with the design allowable heat loads at 100% power.

Section B - PCRV Structural Performance

Data

No changes in PCRV structural performance have been identified since the previous report.

Section C - PCRV Leak Tightness

Data

The primary coolant makeup rate is higher than expected due to a known leak in the Loop 2 steam generator interspace. Purified helium leaks from the interspace into the reheat steam piping. This was explained and reported in RO 50-267/80-30/03-L-0.

TABLE B-3A.1

PCRV SUBSECTION	HEAT LOAD (X 10^6 BTU/HR) 80% POWER		DESIGN ALLOWABLE AT 100%
	LOOP 1	LOOP 2	
Top Penetrations	0.95	0.95	2.270
Core Support Floor Top	1.27	1.33	------
Core Support Floor Side	0.23	0.47	------
Core Support Floor Bottom	0.74	0.94	------
Total Core Support Floor	2.20	2.74	8.076
Upper Barrel and Top Head	1.15	1.16	6.292
Lower Barrel	0.70	0.83	3.350
Bottom Head and Bottom Penetrations	1.22	1.54	4.887
Total by Loop	6.22	7.22	12.435
TOTAL PCRV	13.44		24.87

Section A - PCRV Liner Cooling System

Comparison of Predicted and Actual Data

The heat load to certain areas of the PCRV is higher than expected at the power levels reached so far. Analysis of the data by General Atomic Company indicates no degradation of the PCRV structural performance is to be expected at the anticipated heat loads at 100% power based upon an extrapolation of the present operating data.

Section B - PCRV Structural Performance

Comparison of Predicted and Actual Data

1. No significant changes in tendon load were observed.

2. The recorded values of concrete strain were in general agreement with predicted values.

3. No unanticipated increase in concrete temperature was indicated by embedded thermocouples.

Section C - PCRV Leak Tightness

Comparison of Predicted and Actual Data

Total helium loss from the PCRV and associated piping is greater than expected.

PRIMARY SYSTEM PERFORMANCE (B-4)

Data

Table B-4.1 shows the data collected at 80% power for all four helium circulators.

TABLE B-4.1

Reactor Power	80%
HELIUM CIRCULATOR 1A	XXXXX
Speed (RPM)	7,650
FLow (%)	85
Δ/P (PSID)	63
Diffuser Δ/P (PSID)	4.1
Compressor Δ/P (PSID)	5.7
Steam Turbine Inlet Pressure (PSIG)	700
Steam Turbine Outlet Pressure (PSIG)	460
Steam Turbine Bypass Pressure Ratio	1.54
HELIUM CIRCULATOR 1B	XXXXX
Speed (RPM)	7,600
Flow (%)	89
Δ/P (PSID)	6.6
Diffuser Δ/P (PSID)	3.4
Compressor Δ/P (PSID)	5.6
Steam Turbine Inlet Pressure (PSIG)	700
Steam Turbine Outlet Pressure (PSIG)	460
Steam Turbine Bypass Pressure Ratio	1.54

TABLE B-4.1 (CONTINUED

Reactor Power	80%
HELIUM CIRCULATOR 1C	XXXXX
Speed (RPM)	7,700
Flow (%)	95
Δ/P (PSID)	6.7
Diffuser Δ/P (PSID)	3.8
Compressor Δ/P (PSID)	5.8
Steam Turbine Inlet Pressure (PSIG)	690
Steam Turbine Outlet Pressure (PSIG)	450
Steam Turbine Bypass Pressure Ratio	1.52
HELIUM CIRCULATOR 1D	XXXXX
Speed (RPM)	8,000
Flow (%)	87
Δ/P (PSID)	6.5
Diffuser Δ/P (PSID)	*
Compressor Δ/P (PSID)	5.9
Steam Turbine Inlet Pressure (PSIG)	690
Steam Turbine Outlet Pressure (PSIG)	450
Steam Turbine Bypass Pressure Ratio	1.52
CORE PRESSURE DROP	4.45

*Instrument Out of Service

PLANT INSTRUMENTATION PERFORMANCE (B-5)

Data

No significant new information was revealed by the data collected during the report period.

Comparison of Actual and Predicted Data

The nuclear power level instrumentation is still subject to decalibration as a result of control rod movement as reported in the previous report. Investigation and evaluation of observed response continues as higher power levels are reached and changing control rod patterns are used.

SHIELDING SURVEYS (B-12)

Data

80% power level.

South wall hot service facility - 2.8 mr/hour.

Gas waste vacuum tank - 1.5 mr/hour

All other locations were less than 1 mr/hour. All locations were well within anticipated values.

RADIOCHEMICAL ANALYSIS OF THE PRIMARY COOLANT (B-13)

Data

The measured noble gas activity by isotope at 80% power is shown in Table B-13.1. The calculated release fraction (R/B) for these noble gas isotopes is shown in Table B-13.2.

TABLE B-13.1

MEASURED STEADY STATE ACTIVITIES FOR NOBLE GAS ISOTOPES AT 80% POWER

Power	Activity in Primary Coolant by Isotope ($\mu Ci/cm^3 \times 10^3$ STP)					
	Kr-85m	Kr-88	Kr-89	Xe-135	Xe-137	Xe-138
80%	1.45	2.61	1.34	2.31	1.84	2.78

TABLE B-13.2

MEASURED R/B VALUES FOR NOBLE GAS ISOTOPES AT 80% POWER

Power	$R/B \times 10^6$					
	Kr-85m	Kr-88	Kr-89	Xe-135	Xe-137	Xe-138
80%	8.18	4.39	1.68	4.08	1.64	2.02

THE FORT ST. VRAIN

INITIAL APPROACH TO

POWER TESTS (B-SERIES)

INTERIM REPORT 20

Report for Period Ending August 22, 1981

TABLE OF CONTENTS

INITIAL APPROACH TO POWER TESTS

(B SERIES STARTUP TESTS)

The initial approach to power is accomplished in a series of discrete power level stages. At each power level, tests are made to measure the characteristics of the plant and to ensure that the plant is within its design limits, and the power can be safely increased to the next stage.

The initial phase of the approach-to-power program will increase the reactor power and steam conditions in stages until approximately 28% power when rated steam conditions are achieved. From this level to full power, the reactor power is increased in stages maintaining rated steam conditions. The sequence for the performance of these tests is given in Figure 1, together with the corresponding approximate reactor power levels. The reactor power levels, helium flow rates, feedwater flow rates, steam temperatures, and steam pressure given in the following description of the initial approach-to-power may differ somewhat from those in the actual approach to power due to change in test requirements or improvements in operating methods identified during other tests.

In general, the initial approach-to-power will be accomplished in the following order:

1. Feedwater flow will first be established through both steam generator loops and the bypass flash tank system using a boiler

feedpump. Helium flow through the core will be provided using one circulator in each loop.

2. The reactor power will be increased to approximately 2%.

3. The reactor power, feedwater flow, and helium flow rate will be simultaneously increased to 5% power, 20% helium flow, and 25% feedwater flow using reactor generated steam from the bypass flash tank supplemented by the auxiliary boiler to power the circulating turbines, turbine driven boiler feedpump, and other plant steam requirements.

4. The reactor power will then be increased to approximately 8%, concurrent with an increase in feedwater flow to about 30%. The helium flow will be maintained at about 20% during this power increase. At this condition, the second circulator in each loop will be started, maintaining constant helium flow, and the main steam pressure will be increased to 2,400 psig.

5. The reactor power will be increased to about 11%, and feedwater will be reduced to 25% to initiate boiling.

6. The reactor power will be increased to about 18% simultaneously with an increase in helium flow to about 33%, maintaining 25% feedwater flow, followed by an increase in reactor power to about 26% with a helium flow of 49%. At this condition, the main steam temperature will be about 800 degrees fahrenheit.

7. The helium flow will then be reduced to about 40% concurrent with a slight adjustment of the reactor power to about 28%.

8. The reactor power will be increased in stages to about 40%, 50%, 60%, 80%, and finally to 100% of full power. During these power level increases, the helium flow rate through the core will be increased to maintain full steam conditions.

This report covers tests performed between 70% and approximately 88% reactor power.

Each power level was maintained for a period of time to perform one or more of the following tests. Preliminary analysis of these measurements, as specified in the overall controlling test document, was completed prior to increasing the reactor power to the next stage.

Steam System Performance Tests (B-1)

Just prior to steaming, and at subsequent power levels during the initial rise-to-power, data will be accumulated and analyzed on the performance of the steam generators, the turbine, and the steam plant auxiliaries. Measurements of the turbine performance will be made at the lower power levels, and the turbine will be loaded at about 28% reactor power.

Analysis of Chemical Impurities in the Primary Coolant (B-2)

As the reactor power level is increased to about 11% of rated, the core and reactor internals will experience temperatures in excess of those reached during the core heat-up for reactivity coefficient measurements. At these temperatures, additional impurities will be degassed. Data on the performance of the helium purification system in removing these chemical impurities from the primary coolant will be taken and analyzed.

PCRV Performance Tests (B-3)

As the reactor power level is increased to 28% power, the helium pressure and temperatures approach their quarter load values which results in a system heat load of approximately 80%. At each power level stage up to 28% power, and at selected stages up to full power, data will be taken and analyzed on the performance of the PCRV and its cooling system on the structural response of the PCRV to increased internal pressure and on the primary system helium use rate.

Primary Coolant System Performance Tests (B-4)

At each power level, data on the performance of the helium circulators and their auxiliaries will be taken and analyzed. Measurements of the radial power distribution (region peaking factors) will be made at approximately 2%, 5%, and 8% reactor power. Data on the performance and calibration of the core helium flow orifice valve will be obtained at approximately 28%, 50%, and 100% reactor power.

Plant Instrumentation Performance Tests (B-5)

In these tests, the performance of the portions of the plant instrumentation, which could not be tested prior to power operation, will be checked. The nuclear instrumentation will be calibrated by means of heat balance measurements and analyses. The calibration of the condensate and feedwater flow instrumentation and the core region outlet thermocouples will be checked. The core region outlet thermocouple test will be performed just prior to the first adjustment of the helium flow orifices at approximately 8% power and again at approximately 100% power.

Plant Transient Performance Tests (B-6)

In these tests, the transient performance of the plant will be tested and analyzed. The testing will include: a scram and turbine trip from approximately 28% reactor power with rated steam conditions, a turbine trip from approximately 40% reactor power, a main turbine generator load rejection from approximately 60% reactor power to house load, sequential tripping of the two circulators in a loop from

approximately 80% reactor power and resultant loop shutdown, and boiler feedpump start and stop transients.

Plant Automatic Control System Performance Tests (B-7)

The components of the automatic control system will be placed into service and tested as the controlled variables come into their controllable range. Dynamic verification tests of the control system will be performed at selected power levels during the power level increase of the initial approach to full power. A demonstration of full load change from approximately 100% to approximately 25% turbine load will be made under full automatic control.

Reactor Coefficient Measurements (B-8)

Measurements of changes in reactivity will be made during the approach to full power by measuring the change in control rod positions required to produce a core temperature and reactor power level change.

Differential Control Rod Worth Measurements (B-9)

The reactivity worth of control rods which are moved during the initial rise-to-power will be measured using a reactivity computer to obtain the instantaneous reactivity change produced by a control rod motion.

Xenon Buildup and Decay Measurements (B-10)

The reactivity change produced by buildup, burnout, or decay of xenon poison following a power level change will be measured by recording the change in the critical control rod positions following a change.

Xenon Stability Test (B-11)

In this test, the absence of any sustained xenon oscillations is demonstrated. At 100% power, a perturbation is produced from equilibrium xenon by inserting a control rod in one region and withdrawing a control rod in another region. The indicated power level and region outlet temperatures are recorded as a function of time and analyzed for the presence of any oscillation produced by xenon.

Shielding Surveys (B-12)

At approximately 28% reactor power and approximately 100% reactor power, surveys of the radiation levels within the plant are performed. An additional survey is taken during and following any regeneration of the helium purification system. These measured data are recorded and analyzed to demonstrate the adequacy of the shielding design.

Radiochemical Analysis of the Primary Coolant (B-13)

In this test, the radioactive gaseous fission products in the primary coolant will be sampled and analyzed. These tests are used in the initial startup phase to define fuel fission product release-to-birth ratio at zero burnup and will yield information on the fraction of failed fuel particle coatings. This test is performed at each major power level of the initial rise-to-power.

RISE-TO-POWER TESTING SEQUENCE

REACTOR POWER (% OF RATED)	2%	5%	8%	11%	18%	26%	28%	40%	50%	60%	80%	100%
Steam System Performance			0	0	0	0	0	0	0	0	0	0
Analysis of Chemical Impurities	0	0	0	0	0	0	0	0	0	0	0	0
PCRV Performance	0	0	0	0		0	0	0	0	0	0	0
Primary System Performance	0	0	0	0	0	0	0	0	0	0	0	0
Plant Instrumentation	0	0	0	0	0	0	0	0	0	0	0	0
Plant Transient Performance							0		0	0	0	
Auto Control System	0	0	0	0		0	0	0	0	0	0	0
Reactivity Coefficients	0	0	0	0	0	0	0	0	0	0	0	0
Differential Rod Worths	0	0	0	0	0	0	0	0	0	0	0	0
Xenon Buildup and Decay			0			0			0			0
Xenon Stability												0
Shielding Surveys							0					0
Radiochemical Analysis	0	0	0	0	0	0	0	0	0	0	0	0

FIGURE 1

ACKNOWLEDGEMENT

The contents of this report on the results of B Series Startup Testing at Fort St. Vrain, Unit No. 1, have been taken from unpublished, internal reports of General Atomic Company and Public Service Company of Colorado.

This is an interim report based on preliminary data; and therefore, both data and results are subject to change. This report will be supplemented periodically as further testing is completed.

HISTORICAL SUMMARY OF PLANT OPERATION

The last testing reported was covered in interim report number 8 for the period ending August 22, 1978. After this, the plant continued operation with the Nuclear Regulatory Commission limit of 70% power.

In February, 1979, the shutdown for the first refueling was started with the shutdown extending until June, 1979. After startup, normal operation was resumed until September 1, 1979, when the plant was shutdown due to inconsistencies found in the safety related piping and hangers. This was found during an audit and analysis committed to by the Company and was reported in RO 50-267/79-35/01-T-0.

On September 15, 1979, the reactor was started up and operation continued until October 26, 1979, when a maintenance shutdown began to irstall the region constraint devices (RCD). The RCD installation was completed, and the reactor taken critical on December 25, 1979. The RCD's are metal clamps that join the tops of all the regions together to maintain nearly uniform gap flow areas between regions.

Following a reactor scram, 1B helium circulator static seal failed in January, 1980, requiring the circulator to be replaced with the spare.

On February 17, 1980, the reactor was taken critical again, and the turbine generator wa put on line March 5, 1980.

Operation continued throughout the year and into 1981.

On March 16, 1981, Amendment 23 was issued, which authorized fluctuation testing at greater than 70% power. Functuation testing was started on March 18, 1981, with power levels of 70% to 88% being reached during the period April 17 to April 24, and on May 13, 1981.

On May 13, 1981, the turbine generator tripped on high vibration, due to a loose shroud on the low pressure turbine blading and was followed by a reactor scram. As the turbine repair was to take several weeks, the second refueling out ge was started on May 20, 1981. During this period, 1B circulator was replaced again due to another failed static seal.

The refueling shutdown was finished in July, 1981, and the reactor was taken critical on July 13, 1981. Plant operation has continued throughout the rest of this report period at power levels up to 70% and 230 MWe.

TESTING SUMMARY

This report covers the period from May 23, 1981, through August 22, 1981, and the analysis of the test results collected during the last period of February 23, 1981, to May 22, 1981.

Due to the installation of the region constraint devices and Nuclear Regulatory Commission requirements, the B Series Startup Tests, RT-500K (fluctuation testing at greater than 70% power) and RT-485 (control rod drive temperature data collection) were run concurrently, with RT-500K being the controlling document for testing under normal conditions from 40% to 100% power. The test schedule is called out in T-164 (coordination procedure for testing at greater than 70% power) and is included at the end of the test summary section.

Testing at greater than 70% power was initiated on three occasions (April 17, 1981, April 24, 1981, and May 13, 1981) up to a maximum power level of 88%, at which time turbine vibrations necessitated a plant shutdown and the second refueling outage was begun.

Based on tests completed so far, the following conclusions can be drawn concerning plant behavior (for details refer to individual B Series Test Reports).

1. The steam generator performance at 80% reactor power was acceptable. The maximum measured helium hot streak temperature was less than or equal to 30 degrees fahrenheit about the mixed mean inlet helium temperature, and the maximum main steam subheader deviation within a steam generator module was within the 18 degrees fahrenheit allowable deviation. The maximum measured crossover tube temperature was 849 degrees fahrenheit, which was well below the allowable 950 degrees fahrenheit limit. The average reheat steam outlet temperature was 991 degrees fahrenheit, the temperature controller was set for 990 degrees fahrenheit. The average main steam outlet temperature was 997 degrees fahrenheit. The performance of the main turbine generator at 80% reactor power was acceptable. (SUT B-1)

2. During the performance of this test, variations of the primary system impurity levels were experienced due to problems with continued off-gassing and moisture ingress resulting from circulator trips. Although the helium purification system was in operation, total oxidants were in excess of 10 ppm during the testing periods. See Reportable Occurrence Report No. 50-267/81-027. (SUT B-2)

3. Based upon the PCRV heat load data taken during the test, the maximum heat load relative to the acceptance criteria was experienced in the top penetration area of the PCRV. The heat load in this area was 104% of the acceptance criteria value. The

heat loads in all other areas of the prestressed concrete reactor vessel were lower than design values. This situation is similar to that observed in all previous test data for B-3, Section 2R. (SUT B-3)

4. The maximum PCRV load cell change relative to other corresponding load cells was the top head cross head tendon TIR-M1. However, it is still well above the recommended low limit. It is recommended that load cell TIR-M1 be included in the next scheduled lift off test. The zero psig reference data was obtained nearly two years ago and is most likely no longer valid due to load cell changes and long term PCRV creep. (SUT B-3)

5. Primary coolant makeup rate is higher than expected due to a known leak in the Loop 2 steam generator interspace. Purified helium leaks into the reheat steam piping. This was explained and reported in RO 50-267/80-30/03-L-0. (SUT B-3)

6. The measured values of: circulator pressure rise, circulator helium flow, and core pressure drop continue to deviate from the anticipated values. The measured circulator pressure rise was 40% below predicted, consistent with the previous trends. The helium flow was 9% higher than predicted to match main and reheat steam temperatures, and the core pressure drop was 46% below predicted. After installation of the region constraint devices,

fluctuation testing at greater than 70% reactor power was initiated.

Testing performed to date indicates that the region constraint devices do prevent the onset of fluctuations. However, a single core redistribution has been noticed each time power is increased towards 100%.

Future testing will be done as plant conditions permit. (SUT B-4)

7. The expected decalibration of the nuclear instrumentation as a function of control rod pattern continues to be observed. The nuclear channels are checked daily and at every 10% power level change, and adjusted as required to agree with a heat balance. (SUT B-5)

8. The shielding survey at 80% power showed less than 1 mr/hour in all locations except the south wall of the hot service facility, which read 2.8 mr/hour, and the gas waste vacuum tank which read 1.5 mr/hour. All the locations were well within anticipated. (SUT B-12)

9. The primary coolant radioactivity level and the fission product release to birth ratio (R/B) continue to be less than

anticipated. At 80% power, Ba-139 was observed in the fast gas sample. (SUT B-13)

STEADY STATE B-O DATA COLLECTION*

SUT	TITLE	APPROXIMATE POWER LEVELS ~60	~70	~80	~90	~100
B1-1	Steam Generator Steady State	0	0	0	0	R
B1-3	Steam Generator Steady State (Data with Plugged Tube)			R		
B1-4	Steam Generator Steady State (HP Feedwater Heater Bypassed)			R		
B1-6	Turbine Generator Steady State Performance	0	0	0	0	R
B2-1	Primary Coolant System Impurities	0	0	R	0	R
B2-2	Purification Chemical Impurities	0	0	R	0	R
B2-3	Calibration of Gas Chromatograph	0	0	R	0	R
B3-1R	Liner Cooling Maximum Temperatures	0	0	R	0	R
B3-1S	Liner Cooling ADJ (As Required)	-	-	-	-	-
B3-1Q	PCRV Liner Cooling	0	0	0	0	R
B3-2Q	PCRV Data Scan	0	0	0	0	R
B3-2R	PCRV Internal Temperatures	0	0	R	0	R
B3-3A	PCRV Leak Tightness	0	0	R	0	R
B3-3B	Full Pressure PCRV Leakage SR 5.2.16a-M (as required)	-	-	-	-	R
B4-1(0)	Circulator Primary Coolant	0	0	0	0	R
B5-1	Nuclear Instrument Calibration	0	0	0	0	R
B5-2	Core Region TC Calibration	-	-	-	-	R
B5-3	Feedwater Flow Calibration	0	0	0	0	R
B11	Xenon Stability Test					R
B12-2	Shielding Survey	0	0	0	0	R
B13-1	Radioactive Gas Analysis	0	0	0	0	R
B13-2	Iodine Probe Analysis	-	-	-	-	R
*RT-485	Control Rod Drive Internal Temperatures *(Data to be taken at 6% Intervals Between 70 and 100%)	0	-	-	-	-
RP390-M	Generator Temperature Monitoring	0	0	0	0	0

"O" Optional Data Taking
"R" Required Data Taking

*NOTE: For further details on individual B-O steady state data taking frequency see pages 10-15.

FIGURE 1

STEAM SYSTEM PERFORMANCE VERIFICATION (B-1)

Data

1. Typical steam generator temperature data taken at approximately 80% is shown in Table B-1.1.

2. Typical main turbine parameters at about 80% reactor thermal power (80% turbine load) are shown in Table B-1.2.

TABLE B-1.1

STEAM GENERATOR MODULE TEMPERATURES

Module	Inlet Helium Temperature	Cold Reheat Steam Temperature °F	Reheat Steam Temperature	Main Steam Temperature
B-1-1	1347	546	998	980 (2)(3)
B-1-2	1319	539	980	997 (1)
B-1-3	1346	543	1001	1002
B-1-4	1330	540	990	1002
B-1-5	1340	544	994	1000
B-1-6	1333	541	992	995
B-2-1	1347	526	990	996
B-2-2	1332	527	998	1000
B-2-3	1342	523	991	990 (3)
B-2-4	1357	530	1005	1000 (2)
B-2-5	1340	526	978	1000 (4)
B-2-6	1323	525	985	998 (1)

(1) Low helium temperature to B-1-2 and B-2-6.

(2) High helium temperature to B-2-4 and B-1-1.

(3) Trim on B-1-1 and B-2-3 trim valve needs to be corrected.

(4) Possible high attemperation flow to B-2-5.
Low hot reheat steam at module outlet.

INSTRUMENT	DATA
Linear Reactor Power Level	79%
Wide Range Reactor Power Level	100%
Helium Pressure	665 PSIA
Helium Pressure	660 PSIA
Helium Pressure	680 PSIA
Circulator Inlet Temperature	688°F
Circulator Speed, 1A	7500 RPM
Circulator Speed, 1B	7600 RPM
Circulator Speed, 1C	7600 RPM
Circulator Speed, 1D	7900 RPM
Circulator Helium Flow, 1A	74%
Circulator Helium Flow, 1B	76%
Circulator Helium Flow, 1C	82%
Circulator Helium Flow, 1D	75%
Circulator Helium Differential Pressure, 1A	4.5 PSI
Circulator Helium Differential Pressure, 1B	4.4 PSI
Circulator Helium Differential Pressure, 1C	4.4 PSI
Circulator Helium Differential Pressure, 1D	4.5 PSI
Feedwater Temperature, Loop 1	370°F
Feedwater Temperature, Loop 2	370°F
Feedwater Flow, Loop 1	860,000 lbs/hr
Feedwater Flow, Loop 2	860,000 lbs/hr
Cold Reheat Pressure Before Circulators, Loop 1	695 PSIG
Cold Reheat Pressure Before Circulators, Loop 2	695 PSIG
Cold Reheat Pressure After Circulators, Loop 1	460 PSIG

INSTRUMENT	DATA
Cold Reheat Pressure After Circulators, Loop 2	450 PSIG
Main Steam Pressure, Loop 1	2500 PSIG
Main Steam Pressure, Loop 2	2450 PSIG
Main Steam Temperature, Loop 1	1005°F
Main Steam Temperature, Loop 2	1010°F
Turbine First Stage Pressure	1200 PSIG
Turbine Throttle Pressure	2450 PSIG
Hot Reheat Steam Pressure, Loop 1	440 PSIG
Hot Reheat Steam Pressure, Loop 2	430 PSIG
Desuperheater Water Flow, Loop 1	45,000 lbs/hr
Desuperheater Water Flow, Loop 2	45,000 lbs/hr
Feedwater Valve Differential Pressure, Loop 1	120 PSID
Feedwater Valve Differential Pressure, Loop 2	100 PSID
Hot Reheat Pressure	415 PSIG
Hot Reheat Temperature at Turbine Generator Inlet	986°F
Steam Chest Pressure	2430 PSIG
Flash Tank Pressure	700 PSIG
Turbine Generator Load	270 MW
Module Feedwater Flow, Channel 1	67.0%
Module Feedwater Flow, Channel 2	63.5%
Module Feedwater Flow, Channel 3	66.5%
Module Feedwater Flow, Channel 4	65.5%
Module Feedwater Flow, Channel 5	66.0%
Module Feedwater Flow, Channel 6	66.0%
Module Feedwater Flow, Channel 7	72.5%

INSTRUMENT	DATA
Module Feedwater Flow, Channel 8	66.0%
Module Feedwater Flow, Channel 9	69.0%
Module Feedwater Flow, Channel 10	71.5%
Module Feedwater Flow, Channel 11	68.5%
Module Feedwater Flow, Channel 12	69.5%
Module Main Steam Temperature, Channel 1	980°F
Module Main Steam Temperature, Channel 2	997°F
Module Main Steam Temperature, Channel 3	1002°F
Module Main Steam Temperature, Channel 4	1002°F
Module Main Steam Temperature, Channel 5	1000°F
Module Main Steam Temperature, Channel 6	995°F
Module Main Steam Temperature, Channel 7	996°F
Module Main Steam Temperature, Channel 8	1000°F
Module Main Steam Temperature, Channel 9	999°F
Module Main Steam Temperature, Channel 10	1000°F
Module Main Steam Temperature, Channel 11	1000°F
Module Main Steam Temperature, Channel 12	998°F
Circulator Outlet Steam Temperature, Loop 1	650°F
Circulator Outlet Steam Temperature, Loop 2	632°F
Reheater Inlet Temperature, Channel 1	546°F
Reheater Inlet Temperature, Channel 2	539°F
Reheater Inlet Temperature, Channel 3	543°F
Reheater Inlet Temperature, Channel 4	540°F
Reheater Inlet Temperature, Channel 5	544°F
Reheater Inlet Temperature, Channel 6	541°F

INSTRUMENT	DATA
Reheater Inlet Temperature, Channel 7	526°F
Reheater Inlet Temperature, Channel 8	527°F
Reheater Inlet Temperature, Channel 9	523°F
Reheater Inlet Temperature, Channel 10	530°F
Reheater Inlet Temperature, Channel 11	526°F
Reheater Inlet Temperature, Channel 12	525°F
Reheater Outlet Temperature, Channel 1	998°F
Reheater Outlet Temperature, Channel 2	980°F
Reheater Outlet Temperature, Channel 3	1001°F
Reheater Outlet Temperature, Channel 4	990°F
Reheater Outlet Temperature, Channel 5	994°F
Reheater Outlet Temperature, Channel 6	992°F
Reheater Outlet Temperature, Channel 7	990°F
Reheater Outlet Temperature, Channel 8	988°F
Reheater Outlet Temperature, Channel 9	991°F
Reheater Outlet Temperature, Channel 10	1005°F
Reheater Outlet Temperature, Channel 11	978°F
Reheater Outlet Temperature, Channel 12	985°F

TABLE B-1.2

MAIN TURBINE GENERATOR OPERATING PARAMETERS

Item	Predicted Values	Measured Data
Load MW(e)	268	264
Main Steam Temperature	1,000	990
Reheat Steam Temperature	1,000	1000
First Stage Pressure, PSIG	1,249	1230
Main Steam Pressure	2,400	2390
Cold Reheat Pressure	675	670
Feedwater Temperature	386	375
Condenser Pressure	2.5	2.25
Turbine Speed	3,600	3,650
Attemperation Flow	110,200	90,000
Maximum Vibration (Pt. 5)	---------	3.65 Mils

INSTRUMENT	DATA
Load	264 MW
Number of Hours at Load	2 Hours
Reactive KVA	10 Mega Vars
Hydraulic Piston Stroke (Control Valves)	100%
Hydraulic Piston Stroke (Control Valves)	100%
Hydraulic Piston Stroke (Control Valves)	13%
Hydraulic Piston Stroke (Control Valves)	0%
Steam Chest Pressure	2390
Main Steam Temperature	990°F
First Stage Pressure	1230 PSIG
High Pressure Exhaust to Reheater	670 PSIG
High Pressure Exhaust to Reheater	710°F
Reheat Bowl Pressure	380°F
Hot Reheat Temperature	1000°F
Hot Reheat Steam to IP Turbine Pressure	420 PSIG
Cold Reheat Steam PT #14	690°F
Fifteenth Stage Pressure, Heater 1	6.9
Fourteenth Stage Pressure, Heater 2	11.0
Thirteenth Stage Pressure, Heater 3	20.5
Eleventh Stage Pressure, Heater 4	64
Tenth Stage Pressure, Heater 5	116
Eighth Stage Pressure, Heater 6	185
Feedwater Temperature Leaving Top Heater	375
Exhaust Pressure	23 Hg
Barometer	25.25 Hg

INSTRUMENT	DATA
Steam Flow	1.6×10^6 lbs/hr
Condensate Flow	1.64×10^6 lbs/hr
Quantity of Makeup Demineralizer Water	53 GPM
Feedwater Heaters in Service	All But #5
Quantity of Attemperation to Reheat Loop 2	45×10^3 lbs/hr
Quantity of Attemperation to Reheat Loop 1	45×10^3 lbs/hr
Main Steam Pressure	2430
Main Steam Pressure	2450
Operating Oil Pressure	225
Bearing Header Pressure	25
Main Pump Suction Pressure	23
Turbine Speed	3650
Armature Current	6950
Armature Current	7050
Armature Current	6950
Armature Voltage	2190 KV
Field Current	1700 Amps
Field Voltage	200 Volts
Field Temperature	54°C
Hydrogen Pressure	19 PSIG
Hydrogen Purity	98%
Differential Fan Pressure, Water	3.1
Exhaust Hood Temperature	112
Exhaust Hood Temperature	110
Cold Gas Temperature	32°C

INSTRUMENT	DATA
Cold Gas Temperature	34°C
Cold Gas Temperature	31°C
Cold Gas Temperature	32°C
Stator Temperature, Liquid Out	36°C
Stator Temperature, Liquid In	40.5°C
Stator Temperature, Machine Gas Temperature	37°C
Exciter Air In Temperature	39°C
Exciter Air Out Temperature	28.5°C
Collector Air Out Temperature	38.5°C
Turbine Temperature and Differential Temperature Data Point 1	1010°F
Turbine Temperature and Differential Temperature Data Point 2	1010°F
Turbine Temperature and Differential Temperature Data Point 3	1000°F
Turbine Temperature and Differential Temperature Data Point 4	960°F
Turbine Temperature and Differential Temperature Data Point 5	860°F
Turbine Temperature and Differential Temperature Data Point 6	740°F
Turbine Temperature and Differential Temperature Data Point 7	990°F
Turbine Temperature and Differential Temperature Data Point 8	840°F
Turbine Temperature and Differential Temperature Data Point 9	135°F
Turbine Temperature and Differential Temperature Data Point 10	510°F
Turbine Temperature and Differential Temperature Data Point 11	365°F

INSTRUMENT	DATA
Turbine Temperature and Differential Temperature Data Point 12	160°F
Turbine Temperature and Differential Temperature Data Point 13	710°F
Turbine Temperature and Differential Temperature Data Point 14	710°F
Turbine Temperature and Differential Temperature Data Point 15	690°F
Turbine Temperature and Differential Temperature Data Point 16	700°F
Turbine Temperature and Differential Temperature Data Point 17	860°F
Turbine Temperature and Differential Temperature Data Point 18	840°F
Turbine Temperature and Differential Temperature Data Point 19	130°F
Turbine Temperature and Differential Temperature Data Point 20	60°F
Turbine Temperature and Differential Temperature Data Point 21	55°F
Turbine Temperature and Differential Temperature Data Point 22	140°F
Turbine Vibration Data Point 1	0.12 Mils
Turbine Vibration Data Point 2	0.09 Mils
Turbine Vibration Data Point 3	2.70 Mils
Turbine Vibration Data Point 4	0.92 Mils
Turbine Vibration Data Point 5	3.65 Mils
Turbine Vibration Data Point 6	1.93 Mils
Turbine Vibration "A", #1, Data Point 7	1.3 Mils
Turbine Vibration "A", #2, Data Point 8	.35 Mils
Turbine Vibration "C", #1, Data Point 9	.6 Mils
Turbine Vibration "C", #2, Data Point 10	2.4 Mils

CHEMICAL IMPURITIES IN THE PRIMARY COOLANT (B-2)

Data

Table B-2.1 shows the data collected during operation at greater than 70% power.

TABLE B-2.1

DATE	TIME	CORE OUTLET TEMPERATURE °F	REACTOR POWER %	O_2 PPMV	CO PPMV	CO_2 PPMV	H_2 PPMV	CH_4 PPMV	N_2 PPMV	H_2O PPMV
4-16-81	1600	1353	70	0.0	13.2	3.3	5.6	0.4	28.8	0
	2000	1359	70	0.0	11.8	3.1	8.0	0.3	27.4	0
4-17-81	0200	1363	74	2.3	11.6	2.1	4.8	0.2	27.6	0
	0400	1352	70	0.0	11.8	2.1	5.6	0.4	25.9	0
	1000	1371	74	1.5	12.2	1.8	8.0	0.4	24.5	0
	1200	1378	70	1.5	12.2	1.5	8.0	0.2	24.5	0
	1400	1386	80	0.8	11.7	1.4	8.0	0.4	24.5	0
	1600	1362	74	1.1	11.9	1.7	9.6	0.6	24.5	0
	1800	1364	74	0.0	10.7	1.5	7.2	0.4	67.7	0
	2200	1374	80	0.8	13.0	1.7	8.0	0.2	71.0	0
4-18-81	0200	1371	80	0.0	12.5	1.7	8.0	0.2	52.8	0
	0600	1371	80	1.1	11.2	2.1	7.7	0.3	41.0	0
	0800	1371	80	1.1	12.9	1.7	8.0	0.2	52.8	0
	1200	1372	80	0.0	12.9	1.7	6.4	0.4	40.3	0
	2000	1374	80	0.0	10.2	1.4	7.2	0.2	43.2	0
4-24-81	0900	1369	80	1.0	11.6	1.8	11.1	0.2	71.0	0
	1000	1370	80	1.4	13.1	1.1	12.8	0.2	65.3	0
	1200	1380	84	0.7	13.7	1.2	11.1	0.2	73.0	0
	1400	1366	80	0.7	13.8	1.8	11.1	0.1	55.7	0
	1600	1381	84	2.8	13.5	1.0	11.9	0.4	47.0	0
	1800	1392	88	2.8	17.4	1.8	15.3	0.5	57.6	0
	2000	1392	84	3.2	15.9	1.6	17.0	0.4	51.8	0
	2200	1365	84	2.8	13.5	2.6	11.9	0.3	46.1	0
4-25-81	0000	1364	80	0.7	15.3	1.5	12.8	0.2	72.0	0

PCRV PERFORMANCE TESTS (B-3)

Section A - PCRV Liner Cooling System

Data

The heat loads for each subsection of the PCRV at 80% power are shown in Table B-3A.1 with the design allowable heat loads at 100% power.

Section B - PCRV Structural Performance

Data

No changes in PCRV structural performance have been identified since the previous report.

Section C - PCRV Leak Tightness

Data

The primary coolant makeup rate is higher than expected due to a known leak in the Loop 2 steam generator interspace. Purified helium leaks from the interspace into the reheat steam piping. This was explained and reported in RO 50-267/80-30/03-L-0.

TABLE B-3A.1

PCRV SUBSECTION	HEAT LOAD (X 10^6 BTU/HR) 80% POWER		DESIGN ALLOWABLE AT 100%
	LOOP 1	LOOP 2	
Top Penetrations	0.95	0.95	2.270
Core Support Floor Top	1.27	1.33	------
Core Support Floor Side	0.23	0.47	------
Core Support Floor Bottom	0.74	0.94	------
Total Core Support Floor	2.20	2.74	8.076
Upper Barrel and Top Head	1.15	1 16	6.292
Lower Barrel	0.70	0.83	3.350
Bottom Head and Bottom Penetrations	1.22	1.54	4.887
Total by Loop	6.22	7.22	12.435
TOTAL PCRV	13.44		24.87

Section A - PCRV Liner Cooling System

Comparison of Predicted and Actual Data

The heat load to certain areas of the PCRV is higher than expected at the power levels reached so far. Analysis of the data by General Atomic Company indicates no degradation of the PCRV structural performance is to be expected at the anticipated heat loads at 100% power based upon an extrapolation of the present operating data.

Section B - PCRV Structural Performance

Comparison of Predicted and Actual Data

1. No significant changes in tendon load were observed.

2. The recorded values of concrete strain were in general agreement with predicted values.

3. No unanticipated increase in concrete temperature was indicated by embedded thermocouples.

Section C - PCRV Leak Tightness

Comparison of Predicted and Actual Data

Total helium loss from the PCRV and associated piping is greater than expected.

PRIMARY SYSTEM PERFORMANCE (B-4)

Data

Table B-4.1 shows the data collected at 80% power for all four helium circulators.

TABLE B-4.1

Reactor Power	80%
HELIUM CIRCULATOR 1A	XXXXX
Speed (RPM)	7,650
FLow (%)	85
Δ/P (PSID)	63
Diffuser Δ/P (PSID)	4.1
Compressor Δ/P (PSID)	5.7
Steam Turbine Inlet Pressure (PSIG)	700
Steam Turbine Outlet Pressure (PSIG)	460
Steam Turbine Bypass Pressure Ratio	1.54
HELIUM CIRCULATOR 1B	XXXXX
Speed (RPM)	7,600
Flow (%)	89
Δ/P (PSID)	6.6
Diffuser Δ/P (PSID)	3.4
Compressor Δ/P (PSID)	5.6
Steam Turbine Inlet Pressure (PSIG)	700
Steam Turbine Outlet Pressure (PSIG)	460
Steam Turbine Bypass Pressure Ratio	1.54

TABLE B-4.1 (CONTINUED

Reactor Power	80%
HELIUM CIRCULATOR 1C	XXXXX
Speed (RPM)	7,700
Flow (%)	95
Δ/P (PSID)	6.7
Diffuser Δ/P (PSID)	3.8
Compressor Δ/P (PSID)	5.8
Steam Turbine Inlet Pressure (PSIG)	690
Steam Turbine Outlet Pressure (PSIG)	450
Steam Turbine Bypass Pressure Ratio	1.52
HELIUM CIRCULATOR 1D	XXXXX
Speed (RPM)	8,000
Flow (%)	87
Δ/P (PSID)	6.5
Diffuser Δ/P (PSID)	*
Compressor Δ/P (PSID)	5.9
Steam Turbine Inlet Pressure (PSIG)	690
Steam Turbine Outlet Pressure (PSIG)	450
Steam Turbine Bypass Pressure Ratio	1.52
CORE PRESSURE DROP	4.45

*Instrument Out of Service

PLANT INSTRUMENTATION PERFORMANCE (B-5)

Data

No significant new information was revealed by the data collected during the report period.

Comparison of Actual and Predicted Data

The nuclear power level instrumentation is still subject to decalibration as a result of control rod movement as reported in the previous report. Investigation and evaluation of observed response continues as higher power levels are reached and changing control rod patterns are used.

SHIELDING SURVEYS (B-12)

Data

80% power level.

South wall hot service facility - 2.8 mr/hour.

Gas waste vacuum tank - 1.5 mr/hour

All other locations were less than 1 mr/hour. All locations were well within anticipated values.

RADIOCHEMICAL ANALYSIS OF THE PRIMARY COOLANT (B-13)

Data

The measured noble gas activity by isotope at 80% power is shown in Table B-13.1. The calculated release fraction (R/B) for these noble gas isotopes is shown in Tabla B-13.2.

TABLE B-13.1

MEASURED STEADY STATE ACTIVITIES FOR NOBLE GAS ISOTOPES AT 80% POWER

Power	Activity in Primary Coolant by Isotope ($\mu Ci/cm^3 \times 10^3$ STP)					
	Kr-85m	Kr-88	Kr-89	Xe-135	Xe-137	Xe-138
80%	1.45	2.61	1.34	2.31	1.84	2.78

TABLE B-13.2

MEASURED R/B VALUES FOR NOBLE GAS ISOTOPES AT 80% POWER

Power	$R/B \times 10^6$					
	Kr-85m	Kr-88	Kr-89	Xe-135	Xe-137	Xe-138
80%	8.18	4.39	1.68	4.08	1.64	2.02