Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

November 10, 2016

10 CFR 50.4

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 2
Facility Operating License No. NPF-96
NRC Docket No. 50-391

Subject: **Watts Bar Nuclear Plant, Unit 2, Transmittal of Initial Startup Report to the United States Nuclear Regulatory Commission**

In accordance with the requirements of the Watts Bar Nuclear (WBN) Plant, Dual Unit Updated Final Safety Analysis Report (UFSAR), Chapter 14.2.6, "Test Records," the Tennessee Valley Authority (TVA) is submitting the Initial Startup Report for WBN Unit 2.

The enclosure to this letter provides the Final Startup Report for WBN Unit 2. Initial fuel load, pre-critical testing, initial criticality, and low power physics testing, and power ascension testing are discussed in separate sections of the report. The report details the test objectives, methodology, test results, and problems noted for each of the tests performed.

The test objectives and methodology were developed using the graded approach based on criteria provided in Regulatory Guide (RG) 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2. RG 1.68 was further utilized for the selection of plant structures, systems, and components (SSCs), and design features to be included in the test program. During the power ascension testing program, power ascension tests, surveillance instructions, and other permanent plant tests and technical instructions were performed to demonstrate satisfactory operation of SSCs.

The report addresses test activities and the results of tests performed during the period November 2015 through October 2016. The following table provides a summary of the key WBN Unit 2 milestones and associated dates.

Watts Bar Nuclear Plant, Unit 2 - Milestone Activities	
Milestone	**Date**
WBN Unit 2 Facility Operating License, NPF-96	October 22, 2015
Initial Fuel Load Commencement	December 4, 2015
Initial Criticality	May 23, 2016
Test Plateau, 30% Reactor Thermal Power (RTP)	June 16, 2016
Test Plateau, 50% RTP	July 16, 2016
Test Plateau, 75% RTP	July 29, 2016
Test Plateau, 90% RTP	August 29, 2016
Test Plateau, 100% RTP	October 6, 2016
Commercial Operation	October 19, 2016

The WBN Plant Operations Review Committee has reviewed the report.

There are no new regulatory commitments made in this letter. Please address any questions regarding this submittal to Gordon P. Arent at (423) 365-2004.

Respectfully,

Paul R. Simmons
Site Vice President, Watts Bar Nuclear Plant

Enclosure:

Initial Startup Report to the Nuclear Regulatory Commission,
Facility Operating License No. NPF-96, NRC Docket No. 50-391,
Final Report November 2015 through October 2016

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Watts Bar Nuclear Plant
NRC Project Manager – Watts Bar Nuclear Plant

WATTS BAR NUCLEAR PLANT
UNIT 2
INITIAL STARTUP REPORT
TO THE
UNITED STATES
NUCLEAR REGULATORY COMMISSION

APPROVAL SHEET

Power Ascension Test Manager: ____________________ / 10-21-16

TRG Meeting No. 106

TRG Chairman: ____________________ / 10/21/16

Plant Manager: ____________________ / 10/21/16

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT
UNIT 2

INITIAL STARTUP REPORT
TO THE
UNITED STATES
NUCLEAR REGULATORY COMMISSION

FACILITY OPERATING LICENSE NO. NPF-96
NRC Docket No. 50-391

Final Report
November 2015 through October 2016

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LIST OF ACRONYMS

ADRC	Advanced Digital Reactivity Computer
AFW	Auxiliary Feedwater
AOI	Abnormal Operating Instruction
ARO	All Rods Out
ASME	American Society of Mechanical Engineers
AUX	Auxiliary
BOL	Beginning of Life
BYA	Bypass A (Reactor Trip Breaker A)
BYB	Bypass B (Reactor Trip Breaker B)
CBA	Control Bank A
CBB	Control Bank B
CBC	Control Bank C
CBD	Control Bank D
CERPI	Computer Enhanced Rod Position Indication
CLA	Cold Leg Accumulator
COTS	Channel Operational Tests
CPS	Counts Per Second
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CTL	Chronological Test Log
CV	Concurrent Verification
CVCS	Chemical and Volume Control System
DAS	Data Acquisition System
DCN	Design Change Notice
DCS	Distributed Control System
DRWM	Digital Rod Worth Measurement
eNuPOP	Electronic Nuclear Parameters and Operations Package
FATF	Fuel Assembly Transfer Form
FCV	Flow Control Valve
FHI	Fuel Handling Instruction
FI	Flow Indicator
FIC	Flow Indicating Controller
FW	Feedwater
GO	General Operating Instruction
HFP	Hot Full Power
HS	Hand Switch
HVAC	Heating Ventilation and Air Conditioning
HZP	Hot Zero Power
ICRR	Inverse Count Rate Ratio
ICS	Integrated Computer System

LIST OF ACRONYMS (continued)

INPO	Institute of Nuclear Power Operations
IR	Intermediate Range
ITC	Isothermal Temperature Coefficient
IV	Independent Verification
KBH	Thousand Pounds Per Hour
LC	Level Controller
LCP	Loop Calculation Processor
LCV	Level Control Valve
LPMS	Loose Parts Monitoring System
LPPT	Low Power Physics Test
LVDT	Linear Variable Differential Transformer
M&TE	Measuring and Test Equipment
MCD	Maximum Channel Deviation
MCR	Main Control Room
MED	Maximum Expected Deviation
MFP	Main Feedwater Pump
MFPT	Main Feedwater Pump Turbine
MFW	Main Feedwater
MMI	Man Machine Interface
MPPH	Million Pounds Per Hour
MSIV	Main Steam Isolation Valve
MSV	Main Steam Valve
MTC	Moderator Temperature Coefficient
NI	Nuclear Instrumentation
NIS	Nuclear Instrumentation System
NOB	Nuclear Operating Book
NOTP	Normal Operating Temperature & Pressure
NPG	Nuclear Power Group
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NuPOP	Nuclear Parameters and Operations Package
OPC	Overspeed Protection Controller
OPDP	NPG Standard Department Procedure (Operations)
OPSP	Over Power Setpoint
OTDT	Overtemperature Delta Temperature
OTSP	Over Temperature Setpoint
PAT	Power Ascension Test
PATP	Power Ascension Test Program
PDMS	Power Distribution Monitoring System
PET	Power Escalation Test
PIC	Pressure Indicating Controller

LIST OF ACRONYMS (continued)

PID	Point Identification
PLS	Precautions, Limitations and Setpoints
PMT	Post Maintenance Test
PORC	Plant Operations Review Committee
PR	Power Range
psi	Pounds per square inch
psia	Pounds per square inch absolute
psid	Pounds per square inch differential
psig	Pounds per square inch gauge
PTI	Preoperational Test Instruction
PZR	Pressurizer
QPTR	Quadrant Power Tilt Ratio
RBSS	Rod Bank Select Switch
RCCA	Rod Cluster Control Assembly
RCI	Radiological Control Instruction
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDTC	Rod Drop Test Computer
Reg	Regulating
RHR	Residual Heat Removal
RPI	Rod Position Indicator
RSA	Redundant Sensor Algorithm
RT	Reactor Trip
RTA	Reactor Trip Breaker A
RTB	Reactor Trip Breaker B
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RVLIS	Reactor Vessel Level Instrumentation System
RWP	Radiological Work Permit
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SBA	Shutdown Bank A
SBB	Shutdown Bank B
SBC	Shutdown Bank C
SBD	Shutdown Bank D
SE	Site Engineering
SEQ	Sequence
SFP	Spent Fuel Pool
SG	Steam Generator
SGBD	Steam Generator Blowdown
SI	Surveillance Instruction

LIST OF ACRONYMS (continued)

SOI	System Operating Instruction
SR	Source Range
SRF	Statistical Reliability Factor
SSP	Site Standard Practice
SSPS	Solid State Protection System
SWIF	Seal Water Injection Filter
TE	Temperature Element
TI	Technical Instruction
TR	Temperature Recorder
TRG	Test Review Group
TRI	Technical Requirements Instruction
TTD	Time Trip Delay
TVA	Tennessee Valley Authority
UC	Urgent Change
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
VCT	Volume Control Tank
WBN	Watts Bar Nuclear
WO	Work Order

1.0 INTRODUCTION

The Initial Startup Report for the Watts Bar Unit 2 nuclear plant discusses the results of testing performed from initial core load through full power operation. This report address each of the power ascension tests identified in Chapter 14 of the WBN Unit 2 UFSAR and other license commitments. The report includes a description of the measured values of the operating conditions or characteristics obtained during the testing program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation are also described.

WBN Unit 2 UFSAR Chapter 14.2.6, Test Records, requires the Startup Report be submitted within:

(1) 90 days following completion of the Startup Test Program,
(2) 90 days following resumption or commencement of commercial power operation, or
(3) 9 months following initial criticality, whichever is earliest.

If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

Item (1) is being satisfied since the Power Ascension Test Program was completed on October 6, 2016.

WBN Unit 2 Facility Operating License No. NPF-96 was issued on October 22, 2015. Initial Fuel load commenced with movement of the first fuel assembly at 20:49 on December 4, 2015. Core loading was completed at 02:10 on December 8, 2015. Initial criticality was achieved at 02:16 on May 23, 2016. Further testing was successfully completed at the following plateaus:

Test Plateau, % RTP	Date Completed
30	June 16, 2016
50	July 16, 2016
75	July 29, 2016
90	August 29, 2016
100	October 6, 2016

Initial Fuel load, precritical testing, initial criticality and low power physics testing, and power ascension testing are discussed in separate sections of the report. The report details the test objectives, methodology, test results, and problems noted for each of the tests performed.

1.0 **INTRODUCTION** (continued)

Acceptance Criteria is defined as safety related performance parameters defined in the Design Output, vendor documents, TVA or vendor drawings, NRC commitments, other licensing and design documents, and so forth, that must be exhibited during the performance of a PAT or PET. Failure to meet an acceptance criterion is considered to be a safety related issue.
A 10CFR50.59 Evaluation per NPG-SPP-09.4, 10 CFR 50.59 Evaluations of Changes, Tests, and Experiments or a Technical Evaluation per NPG-SPP-09.3, Plant Modifications and Engineering Change Control, may be required and testing may be stopped. The subsequent course of action will be determined by the nature of the discrepancy and applicable Technical Specifications. Failure to meet Acceptance Criteria will be documented in a Condition Report (CR).

Review Criteria encompasses other performance parameters defined in the UFSAR, design criteria, vendor documents, drawings (TVA or vendor), other licensing, design, setpoint and operational documents that are expected to be exhibited during performance of a PAT or PET. These criteria should be viewed as a guide to possible measurement or design errors. Failure to meet these criteria do not by themselves constitute problems. While prudent measures should be taken to resolve any conflict between measurements and predictions, failure of these criteria do not require 10CFR50.59 Evaluations, per NPG-SPP-09.4, and do not require testing or power ascension to be stopped for resolution. Failure to meet these review criteria will be documented in a CR.

2.0 POWER ASCENSION TEST PROGRAM (PATP) OVERVIEW

The PATP was developed from testing described in Chapter 14 of the WBN Unit 2 UFSAR and requirements specified in Regulatory Guide 1.68, Revision 2, "Initial Test Programs for Water-Cooled Nuclear Power Plants". Testing of the NSSS followed Westinghouse test methodology.

2.1 Administration of the Program

The Site Vice President had the overall responsibility for the PATP.

Overall management of the PATP was directed by the Plant Manager who was responsible for:

- Development and implementation of the PATP to ensure the PATP was conducted in a safe and efficient manner while complying with license provisions and other commitments.
- Establishing the Power Ascension Testing Organization.
- Advising senior management on PATP activities.
- Establishing a Technical Review Group (TRG) as a subcommittee of the PORC to review PATP activities.
- Providing final approval of Power Ascension Tests (PATs) and selected other procedures.
- Ensuring the PATP was conducted in accordance with applicable WBN Administrative Procedures.
- Providing approval to proceed to the next PATP test plateau.
- Providing final approval of all each test package and the Startup Report.

The Power Ascension Test Manager was responsible for:

- Notifying the plant manager of major problems and of the completion of each major test phase (i.e., test sequence) of the program.
- Ensuring the PATs and the PETs were available for NRC review a minimum of 60 days before the scheduled fuel load date.
- Ensuring the technical justification and schedule, including power level for completion of delaying preoperational tests, were provided to the NRC staff prior to fuel load.

2.1 Administration of the Program (continued)

- Ensuring the requirements of TVA-SPP-30.010, Initial Synchronization of TVA Generating Assets to TVA's Transmission System, were met.
- Developing and implementing plans and schedules for the PATP.
- Ensuring testing activities, including planning and scheduling, resulted in safe plant operations and that were not dependent on the performance of untested systems.
- Coordinating and directing overall PATP testing and related activities and requirements with appropriate support groups.
- Supervising test personnel assigned to the power ascension testing group.
- Assigning responsibilities to organizations for specific testing requirements.
- Participating in the review activities of the TRG, and acting as Chairman of the TRG.
- Ensuring the test procedures were reviewed by the TRG.
- Ensuring the Startup Report was reviewed by the TRG.
- Ensuring additional Startup Reports were prepared, reviewed, approved and transmitted to the NRC as needed.
- Ensuring the post-performance test results (i.e., test packages) were reviewed by TRG.
- Ensuring test directors for the PATP were qualified, and met the minimum qualifications of Item 1 and either Item 2 or Item 3 below and ensuring other required individuals (e.g. Independent Verifiers (IV) and Concurrent Verifiers (CV) were qualified to perform the tasks assigned:

    1. Knowledgeable of the test program administration, the system design and operational requirements, and expected plant operational characteristics during the test, and

       Trained as test coordinators in accordance with NPG-SPP-06.9.1, Conduct of Testing.

2.1 Administration of the Program (continued)

2. Possessed a bachelor degree in engineering or physical science, and

   Had two years experience in power plant testing or operation. Included in the two years was one year nuclear power plant testing, operating or training on a nuclear facility.

3. Possessed a high school diploma or equivalent, and

   Had five years experience in power plant testing. Included in the five years were two years of nuclear power plant experience. Credit for up to two years of related technical experience could be substituted for experience on a one-for-one basis.

Technical and administrative oversight of the PATP was performed by TRG which was composed of one representative, or their alternates, from each of the following organizations:

- Plant Operations
- Reactor Engineering
- Site Engineering
- Corporate Nuclear Fuels
- Power Ascension Testing
- Westinghouse

TRG was charged with reviewing PATP testing activities for technical adequacy and affect/impact on nuclear safety, and advising PORC and the plant manager on the disposition of those items reviewed. The responsibilities of TRG included final review and recommendation of approval of all PATP test procedures, revisions, and test results.

Following completion of testing at each major test sequence of the PATP, test results were reviewed by TRG to ensure required tests had been performed. TRG also ensured Acceptance Criteria were satisfied; test deficiencies had proper dispositions, appropriate retesting had been completed, and test results had been reviewed by appropriate designated personnel prior to proceeding to the next major test sequence. This review ensured that all required systems were operating properly and that testing for the next major test sequence could be conducted in a safe and efficient manner.

2.2 Implementation of the Program

The WBN PATP utilized information gained from operating and testing experience at other nuclear plants. This information was used in the development of the PATP test procedures and schedules and to alert personnel to potential problem areas. Test procedures were developed utilizing information obtained from Operating Experience (OE) database. The TVA Operating Experience Program identifies and evaluates experience gained from other TVA nuclear plants, INPO, NRC, equipment suppliers, and from other utilities. Significant operational experience and events were reviewed and integrated into appropriate PATP test procedures to ensure nuclear safety and reliability. To the extent practical, simulator-based training and trial use of the PATP test procedures were performed on the WBN simulator to familiarize personnel with systems and plant operation and to assure technical adequacy of the procedures under simulated plant conditions prior to field use during power operation.

The testing program was conducted by qualified personnel using approved plant administrative, test, and operating procedures. The plant was taken from core load to full power in a highly controlled, conservative, and documented manner which demonstrated, where practical:

- The plant is ready to operate in a manner which will not endanger the health and safety of the public.
- The plant has been properly constructed, and plant performance is satisfactory in terms of established design criteria.
- The plant meets licensing requirements and provides assurance of plant reliability for operation.
- The plant is capable of withstanding anticipated transients and postulated accidents.

The PATP was specified in seven PAT sequence procedures:

- 2-PAT-2.0, Initial Core Loading Sequence
- 2-PAT-3.0, Post Core Loading Precritical Test Sequence
- 2-PAT-4.0, Initial Criticality and Low Power Test Sequence
- 2-PAT-5.0, Test Sequence for 30% Plateau
- 2-PAT-6.0, Test Sequence for 50% Plateau
- 2-PAT-7.0, Test Sequence for 75% Plateau
- 2-PAT-8.0, Test Sequence for 100% Plateau

2.2 Implementation of the Program (continued)

Each PAT sequence procedure called out the performance of other PATs, as well as other designated plant procedures such as PETs, SIs, TRIs, TIs, RCIs and FHIs. The sequence procedures specified the logical performance of required tests and procedures through each test plateau. The sequence procedures also specified general prerequisites, precautions and limitations, and additional operational steps at each test plateau. The detailed test and normal plant procedures called out by the sequence procedures defined step-by-step actions, specific prerequisites and limitations, signoffs, data taking requirements, and test acceptance and review criteria.

The PATP commenced with the receipt of the Facility Operating License on October 22, 2015, and progressed with core loading, precritical testing, initial criticality and low power physics testing, and power ascension testing. Core load procedures directed the initial core load in a prescribed manner which ensured core loading was accomplished in a safe and orderly fashion. Precritical testing brought the plant to hot standby conditions, made measurements, and demonstrated that the plant was ready for critical operation. Initial criticality on May 23, 2016, brought the Unit 2 reactor critical for the first time. Low power physics testing performed measurements on the critical reactor to demonstrate conformance with design predictions prior to power operation. PAT brought the plant to full power, made minor plant instrumentation adjustments, and demonstrated the plant's ability to withstand selected transients. Figure 2.0-1 depicts the time line for the PATP.

Plant events not directly associated with the PATP added to the duration of the program. These events are included in the chronology.

2.3 SUMMARY

Watts Bar Nuclear Plant's Unit 2 Power Ascension Test Program began with the receipt of the operating license. The program encompasses; those preoperational type tests that were deferred to the PATP, the prerequisites required to load the initial core, the initial core loading itself, post core loading tests, initial criticality, low power tests, and at-power tests.

Completion of these tests verified that the unit was properly designed, constructed, and ready to operate in a manner that will not endanger the health and safety of the public, meets contractual and licensing requirements, and provides assurance of plant reliability for operation. The PATP used Regulatory Guide 1.68, Revision 2 and the WBN UFSAR for development of the test requirements.

Issues identified during the testing were resolved except for the following open items that will be resolved by the Corrective Action Program.

OPEN ITEMS:

(1) CR 1208694 initiated WO 118122821 to design and install bracing on multiple Main Steam Traps as a result of visual inspection during 2-PAT-1.4, Pipe Vibration Monitoring.

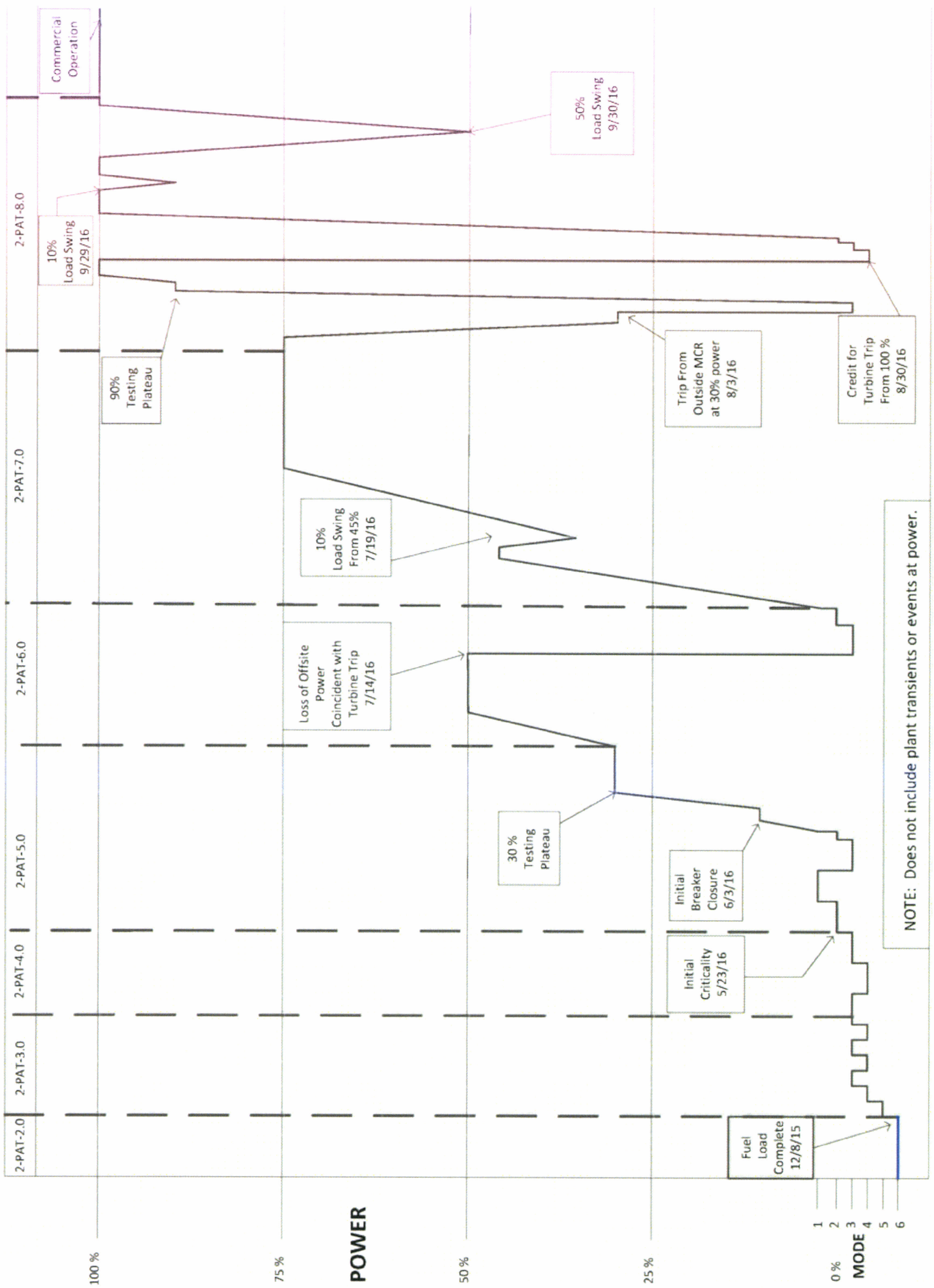
(2) UFSAR Table 14.2 2, Sheet 5 Test Method, refers to an evaluation of thermal expansion at final ambient conditions. This final ambient condition evaluation will be performed later and is tracked by Commitment 118008175.

(3) CR 1208178 was initiated for 2-PT-1-81 being unavailable during testing 2-PAT-1.6, Startup Adjustments of Reactor Control System, and will be repaired by WO 118121693.

(4) CR 1171424 was written during the performance of 2-PAT-1.5, Loose Parts Monitoring System, for three channels taken out of service due to issues. These channels will be repaired under the following work orders.
   - WO 117845593 - Channel 101 Experiencing excessive noise and is alarming due to "IITA" rattling.
   - WO 117843208 - Channel 102 Accelerometer found damaged
   - WO 117843209 - Channel 110 Suspect preamplifier

FIGURE 2.0-1

WBN POWER ASCENSION TEST PROGRAM SCHEDULE OVERVIEW



3.0 WATTS BAR UNIT 2 STARTUP CHRONOLOGY

Note: Power Ascension Tests may be performed at multiple testing plateaus. The description of the individual PAT is documented in the section (plateau) in which it was completed.

10/22/15	-Receipt of WBN Unit 2 Facility Operating License No. NPF-96 from the NRC.
11/19/15	-2-PAT-2.0, Initial Core Load Sequence, was begun. -2-PAT-2.1, Reactor System Sampling for Core Load, was initiated.
11/19/15	-Commenced movement of new fuel into the Spent Fuel Pool in preparation for loading Unit 2 core.
11/22/15	-All 193 fuel assemblies required for Unit 2 fuel load moved into the Spent Fuel Pool.
11/23/15	-Completed Spent Fuel Pool verification for Unit 2 Cycle 1, fuel assembly and component insert verification with no discrepancies noted.
12/3/15	-RCI-159, Radiation Baseline Surveys, pre-fuel load surveys were completed. No Acceptance or Review Criteria is associated with this procedure.
12/4/15	-2-PAT-2.1, Reactor System Sampling for Core Load, completed with all criteria met. - Unit 2 entered Mode 6 at 20:04. -Initiated transport of first fuel assembly to Unit 2 vessel at 20:49.
12/5/15	-Fuel movement was delayed due to issues with refueling machine at 06:23.
12/5/15	-2-PAT-2.2, Response Check Of Core Load Instrumentation After 8 Hour Delay In Fuel Movement, completed at 11:43 with all criteria met and fuel movement resumed.
12/5/15	-Debris was reported on the bottom of the fourth fuel assembly. Debris was removed and fuel movement resumed after appropriate approvals. CR 1112204 was initiated.
12/6/15	-Fuel movement was suspended again due to debris on a fuel assembly. Debris was cleared and fuel movement resumed.
12/8/15	-Initial fuel load for Unit 2 completed at 02:10. -2-PET-105, Initial Core Loading completed. -2-TI-28, Verification Of Core Load Prior To Vessel Closure, was completed at 12:29 with all criteria met.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

12/9/15	-2-PAT-2.0, Initial Core Load Sequence, was completed and TRG approved. -2-PAT-3.0, Post Core Loading and Precritical Test Sequence, pre-requisites were initiated.
12/10/15	-Unit entered Mode 5, maintaining <105°F in RCS in preparation of PAT at the Ambient Plateau.
12/12/15	-RCI-159, Radiation Baseline Surveys - Post Fuel Load Survey was field work complete for applicable Ambient Plateau sections. No Acceptance or Review Criteria were associated with this procedure.
12/16/15	-2-PAT-1.8, Thermal Expansion of Piping Systems, was field work complete for applicable Ambient Plateau sections with all criteria met.
12/23/15	-2-PAT-1.4, Pipe Vibration Monitoring, Section 6.5.6, Condensate - Short Cycle, field work complete with all Acceptance Criteria met. There was no Review Criteria for this test.
1/16/16	-2-PAT-5.1, Dynamic Automatic Steam Dump Control, field work complete for applicable Ambient Plateau sections. There was no Acceptance or Review Criteria for these sections.
1/20/16	-2-PAT-3.10, Reactor Trip System, field work complete with all criteria met.
1/24/16	-2-PAT-3.1, Control Rod Drive Mechanism and CERPI Initial Calibration, field work complete with all Acceptance Criteria met after evaluation. CR 1128950 was written for high current amplitudes and closed following Westinghouse evaluation that determined the measurements to be acceptable. There was no Review Criteria for this PAT. -2-PAT-3.8, Rod Drop Time Measurement and Stationary Gripper Release Timing, Mode 5 Performance, field work complete for applicable Ambient Plateau sections with all Acceptance Criteria met. There was no Review Criteria for this test.
1/26/16	-PAT testing on the plant primary side was suspended on 1/26/16 until plant conditions allowed further testing.
2/1/16	-WO 112989715 Complete for WINCISE Site Acceptance Test (WNA-TP-02985-WBT). This WO satisfied UFSAR Table 14.2-2, Sheet 12, Incore Instrumentation System Test Summary, Acceptance Criteria 1. -2-PAT-1.4, Pipe Vibration Monitoring, field work complete for Section 6.5.7 Condensate - Long Cycle with all criteria met.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

3/19/16	-Unit entered Mode 4, RCS temperature >200°F and <350°F to allow PAT at the 250°F Plateau.
3/21/16	-2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for 250°F Plateau applicable sections with all criteria met.
3/24/16	-RCS temperature increased to 300°F to facilitate PAT.
3/25/16	-2-PAT-1.11, RVLIS Performance Test, field work complete for applicable 300°F Plateau sections with all criteria met.
3/30/16	-Unit entered Mode 3, RCS temperature ≥ 350°F to allow further Power Ascension Testing at the 360°F Plateau.
3/31/16	-2-PAT-1.11, RVLIS Performance Test, field work complete for applicable 360°F Plateau sections with all criteria met. -2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable 360°F Plateau section with all criteria met. -2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete for applicable 360°F Plateau sections with all criteria acceptable for continued heat-up.
4/1/16	-Initiated plant heat-up to 400°F for PAT at 01:14. -2-PAT-1.11, RVLIS Performance Test, field work complete for applicable 400°F Plateau sections with Review Criteria not met. CR 1156425 was written. -2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable 400°F Plateau section with all criteria met. -Increased RCS temperature to 450°F for testing. -2-PAT-1.11, RVLIS Performance Test, field work complete for applicable 450°F Plateau, with all criteria met. -2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable 450°F Plateau sections with all criteria met. -2-PAT-1.8, Thermal Expansion of Piping Systems - field work complete for applicable 450°F Plateau sections. CR's were initiated within the test for seven snubbers not performing as expected. Results indicated no issue with snubbers and approved to continue to next plateau testing. (See Problem Report #1 of 2-PAT-1.8)

3.0 Watts Bar Unit 2 Startup Chronology (continued)

4/2/16	-Unit cool down initiated to repair a check valve with excessive leakage. Additionally, two RCS RTD's were replaced. -Unit entered Mode 4, RCS temperature >200°F and <350°F, at 06:29.
4/8/16	-Unit re-entered Mode 3, RCS temperature ≥ 350°F.
4/10/16	-RCS temperature increased to 500°F. -2-PAT-1.11, RVLIS Performance Test, field work complete for applicable 500°F Plateau sections with all criteria met.
4/12/16	-2-PET-102, Pre-Power Escalation NIS Calibration Data, completed with all criteria met.
4/13/16	-RCS temperature increase to 557°F. -2-PAT-1.11, RVLIS Performance Test - field work complete for applicable section of 557°F data taking only with all criteria met for steady state data collection. -2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for Steady State Data Collection, section 6.7, with all criteria met. -2-PAT-1.4, Pipe Vibration Monitoring - Section 6.5.2, Main Feedwater Pump 2A Start and Steady State Operation on Recirc., was field work complete on 4/13/16 with velocity and displacement Acceptance Criteria not met. CR 1161783 was initiated for an engineering evaluation which concluded equipment was acceptable as is. There was no Review Criteria for this test.
4/14/16	-RCS at normal operating pressure. -2-PAT-1.4, Pipe Vibration Monitoring, completed for Section 6.5.1, Pressurizer Surge, Mode 3, with all criteria met for that section. -2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete for applicable 557°F Plateau sections with an issue outside containment on Protective Device PD07-2. WO 117755755 was to resolve the issue with PD07-2 and investigate any possible issues with PD07-1. Additionally, other components did not move as expected and were evaluated and concluded to be within their working range. (See Problem Reports #2, #3 of 2-PAT-1.8).
4/16/16	-2-PAT-3.2, Pressurizer Spray Capability and Continuous Spray Flow Setting, Section 6.1, Adjustment of the Pressurizer Manual Spray Bypass Valves, was completed. All Acceptance Criteria was met. Review Criteria for MCR alarms was not met with CRs 1161382 and 1160969 written. A Westinghouse evaluation determined the PAT met the operability and design requirements of the pressurizer spray system.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

4/17/16	-Unit was placed in Mode 4 for repairs to the Turbine Driven Auxiliary Feedwater Pump and replacement of PD07-2 shim determined to require adjustment.
5/1/16	-Unit 2 in Mode 3 at 17:36.
5/2/16	-Unit 2 at NOTP at 23:00.
5/3/16	-2-PAT-3.3, RCS Flow Measurement, field work complete with all criteria met.
5/4/16	-2-PAT-1.6, Startup Adjustments of Reactor Control System, field work complete for Mode 3. This performance was data taking only.
5/5/16	-2-PAT-1.4, Piping Vibration Monitoring, Section 6.5.3, Main Feedwater Pump 2B Start and Steady State Operation on Recirc., was field work complete with steady state velocity and displacement exceeding the Acceptance Criteria. CR 1168287 was written for an engineering evaluation and resulted in adjustment of a loose hanger and a retest. The retest was completed on 5/13/16 with satisfactory results. There was no Review Criteria for this test.
5/6/16	-2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, field work complete with all Acceptance Criteria met. One Review Criteria was not met and CR 1168641 was initiated.
5/7/16	-2-PAT-3.0, Attachment 1, field work complete with all Acceptance Criteria met. CR 1168487 was written to document alternate charging flow was not within anticipated range, however, it had no affect on the test acceptance. -2-PAT-3.11, Adjustment of Steam Flow Transmitters at Minimal Flow, field work complete with all Review Criteria met. There was no Acceptance Criteria associated with this performance.
5/8/16	-2-PAT-1.11 - RVLIS Performance Test, Section 6.1.3, field work complete and results indicated Acceptance Criteria would not be met. The system was updated with the new RVLIS constants supplied by Westinghouse to correct the abnormality. CR 1171130 was initiated. -2-PAT-1.12, Common Q Post Accident Monitoring System, Section 6.8, Pump Contact Data Collection at 557°F, was completed with all criteria met. -2-PAT-3.7, Reactor Coolant Flow Coastdown, field work complete with all criteria met.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

5/11/16	-2-PAT-3.8 Rod Drop Time Measurement and Stationary Gripper Release Timing (Mode 3), field work complete. CR 1169659 written for two rods failing a two sigma statistical evaluation. Additional rod drops were performed and the Acceptance Criteria was met. There was no Review Criteria.
5/12/16	-2-PAT-1.4, Pipe Vibration Monitoring, Section 6.5.4, Turbine Bypass Valve 2-FCV-1-105 Transient, was completed with all criteria met. Section 6.5.5, Turbine Bypass Valve 2-FCV-1-111 Transient was completed on 5/12/16 and re-tested on 5/13/16. Engineering evaluation of the retest indicated satisfactory results. CR 1170319 documented the engineering evaluation. -2-PAT-5.1, Dynamic Automatic Steam Dump Control, field work complete with all criteria met after a volume booster adjustment with CR 1170159 and a retest on 2-FCV-1-108.
5/13/16	-2-PAT-3.4, Rod Control and Rod Position Indication (CERPI) - field work complete. The Acceptance Criteria was not met in Sections 6.4 and 6.10. CRs 1168845, 1168881, and 1169602 were written to document failure to meet criteria. The criteria was re-evaluated and it was determined the Acceptance Criteria should be changed to require each Rod Position Indication to indicate rod motion consistent with the group demand indication for the full range of rod travel. A change to the Westinghouse Acceptance Criteria and SAR Change Package No. U2-019 were approved and an urgent change to the procedure incorporated the revised Acceptance Criteria. All Acceptance Criteria were then met.
5/15/16	-2-PAT-3.0, Post Core Loading and Precritical Test Sequence, TRG approved. -2-PAT-4.0, Initial Criticality and Low Power Test Sequence, in progress.
5/16/16	-A cool down to 360°F was initiated to replace a failed RTD on RCS Loop 3 Hot Leg. The unit was stabilized between 355-365°F at 22:59.
5/18/16	-Unit was placed in Mode 4 at 23:58 to facilitate repairs to the Solid State Protection System (SSPS).
5/20/16	-Unit was returned to Mode 3 at 04:15.
5/21/16	-Unit reached NOTP at 01:00. Response time testing of the replaced RCS RTD indicated it did not meet its Acceptance Criteria. A DCN was initiated to revise the Acceptance Criteria to allow entry into Mode 2.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

5/23/16	-Unit entered Mode 2 at 01:04. -Initial criticality at 02:16. -2-PET-201, Initial Criticality and Low Power Physics Testing, completed with all criteria met. -2-PET-103, Reactivity Computer (ADRC), completed with all criteria met. -2-PET-304, Operational Alignment of NIS, applicable sections completed with all criteria met.
5/24/16	-2-PAT-1.5, Loose Parts Monitoring System, was completed with all criteria met. CR 1171424 was written to document three channels removed from service. -2-PAT-1.10, Integrated Computer System (ICS), applicable sections completed with all criteria met. CR 1173586 was initiated for ICS PID quality on several points but did not affect this plateau performance. CR 1174334 was initiated for exceeding the MED between T0457A MCR indicator 2-TI-62-29, RCP 3 LWR RADIAL BRG Temp. -2-PAT-4.0, Initial Criticality and Low Power Test Sequence TRG approved. -2-PAT-5.0 Test Sequence for 30% Plateau in progress.
5/25/16	-Unit 2 entered Mode 1 at 03:33.
5/26/16	-2-PAT-5.3, Automatic Steam Generator Level Control Transients at Low Power, completed with all criteria met.
5/27/16	-2-PAT-5.1, Dynamic Automatic Steam Dump Control, completed Sections 6.3, 6.4 and 6.5 with all Acceptance Criteria met after a procedure and UFSAR revision per Westinghouse Letter LTR-SCS-16-23. -With reactor power between 13 and 14 percent the turbine was rolled for testing in preparations for initial generator synchronization. During the roll up an unanticipated noise was heard and the roll was terminated. A second roll was made later in the evening with similar results. A decision was made to place the Unit in Mode 3 for turbine repairs.
5/28/16	-Unit 2 re-entered Mode 3 at 01:54 after a manual reactor trip for turbine repairs.
5/31/16	-Unit 2 re-entered Mode 2 at 12:00. -Reactor taken critical at 13:39. -Unit 2 entered Mode 1 at 17:49. -RCI-159, Radiation Baseline Surveys, completed. No Acceptance or Review Criteria were associated with this procedure.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

6/3/16	-Unit 2 was synchronized to the grid at 20:39 and holding at 15 percent power to repair steam leaks.
6/4/16	-Unit 2 turbine was manually tripped due to a non-isolable steam leak.
6/5/16	-Unit 2 synchronized at 11:40. -Unit 2 received an automatic reactor trip with a safety injection due to #1 governor valve failing open causing a steam line pressure decrease and subsequent Reactor Trip and Safety Injection at 12:27. Unit was stabilized in Mode 3 following Reactor Trip.
6/8/16	-Unit 2 in Mode 2 at 01:39 after repairs to the governor valve. -Entry into Mode 1 was at 09:32.
6/9/16	-Unit 2 synchronized to grid at 06:40. -Turbine manually tripped due to an non-isolable steam leak at 17:52.
6/11/16	-Unit 2 synchronized to the grid at 13:23 and power increase initiated to the 30% testing plateau.
6/13/16	-2-PAT-1.5, Loose Parts Monitoring System, was completed with all criteria met. CR 1171424 documents three channels removed from service. -2-PAT-1.12, Common Q Post Accident Monitoring System, applicable sections were completed with all criteria met. -2-PAT-1.11, RVLIS Performance Test, applicable sections were completed with all criteria met.
6/14/16	-Completed the initial Flux Map in accordance with 2-TI-41, Incore Flux Mapping, and 2-SI-0-20, Hot Channel Factors Determination. -2-PAT-1.10, Integrated Computer System (ICS), was completed with all criteria met. CR 1181784 was written to address a database error but did not affect this plateau performance.
6/15/16	-2-PAT-1.4, Pipe Vibration Monitoring, completed with all criteria met for observations at the 30% Plateau. -2-PAT-5.3, Automatic Steam Generator Level Control Transients at Low Power, was completed with all Acceptance Criteria met. CR 1181278 was initiated to document one Review Criteria not met. An engineering evaluation determined this did not affect the performance of the test nor invalidate any of the test results and testing should proceed to the next plateau.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

6/15/16	-2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, was completed with all Acceptance Criteria met Two Review Criteria concerning parameters related to Delta T failed. The OTDT calculated by Eagle-21 and provided by the MMI carts indicated approximately 158% and the MCR indicators maximum value is 150%. It was expected the reading from Eagle-21 was accurate and the MCR meters were ranged such that they cannot read the higher value. Additional data was taken at higher power ranges and the meters came on scale with no issue. CR 118246 was written. -2-PAT-5.4 Calibration of Steam and Feedwater Flow Instruments at 30% Power was completed with all criteria met. -2-PAT-1.6 Startup Adjustments of Reactor Control System, was completed. This was data taking only with no Review or Acceptance Criteria at this plateau. -2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete with all criteria met.
6/16/16	-RCI-159, Radiation Baseline Surveys completed. No Acceptance or Review Criteria were associated with this procedure. -2-PAT-5.0, Test Sequence for 30% Plateau, was TRG approved.
6/17/16	-2-PAT-6.0, Test sequence for 50% Plateau, performance section was entered and power increase to 50% Plateau level initiated.
6/20/16	-U-2 turbine tripped due to loss of 2B Main Feedwater Pump from loss of MFP condenser vacuum with a subsequent automatic reactor trip as a result of the S/Gs reaching their low-low trip setpoint. The plant was stabilized in Mode 3.
6/23/16	-U-2 re-entered Mode 2.
6/24/16	-U-2 re-entered Mode 1 and synchronized to the grid.
6/26/16	-U-2 manually tripped the turbine due to a steam leak. Mode 2 was entered and subsequently the reactor was tripped and the unit stabilized in Mode 3.
7/2/16	-U-2 again entered Mode 2 and reactor critical at 03:20. -Unit entered Mode 1 at 07:57. -U-2 synchronized to the grid at 13:36.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

7/7/16	-U-2 reached 50% Plateau power level requirements for testing. -2-PAT-1.5, Loose Parts Monitoring System, was completed with all criteria met. CR 1171424 documents three channels removed from service. -2-PAT-1.11, RVLIS Performance Test, applicable sections completed with all criteria met. -2-PAT-1.12, Common Q Post Accident Monitoring System, applicable sections were completed with all criteria met.
7/8/16	-2-PAT-1.4, Pipe Vibration Monitoring, completed with all criteria met. -2-PAT-1.6, Startup Adjustments of Reactor Control System, completed. This performance was data taking only with no Acceptance or Review Criteria at this plateau. -2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, applicable sections completed with all criteria met. -2-PAT-1.8, Thermal Expansion of Piping Systems, completed with two issues being referred to Site Engineering for evaluation. Engineering review indicated it was acceptable to continue Power Ascension Testing. (See Problem Report 4 of 2-PAT-1.8). -2-PAT-1.10, Integrated Computer System (ICS), completed with all criteria met. -2-PAT-6.3, Calibration of Steam and Feedwater Flow Instruments, at 50% Power completed with all criteria met.
7/9/16	-2-PAT-3.3, RCS Flow Measurement, completed with all criteria met.
7/13/16	-2-PAT-6.1, Automatic Reactor Control System, completed with all criteria met.
7/14/16	-2-PAT-5.2, Turbine Generator Trip With Coincident Loss of Offsite Power Test, completed with all Acceptance Criteria met. CR 1192287 was written to document Tcold decreasing below the 547°F Review Criteria. -Unit 2 entered Mode 3. -2-PAT-1.4, Pipe Vibration Monitoring, applicable section for transient testing was completed with all criteria met.
7/16/16	-2-PAT-6.2, Automatic Steam Generator Level Control Transients completed with all criteria met. -2-PAT-6.0, Test Sequence for 50% Plateau, was approved by TRG.
7/17/16	-Unit 2 re-entered Mode 2 after a planned trip with 2-PAT-5.2, Turbine Generator Trip Coincident With Loss of Offsite Power Test.
7/18/16	-Unit entered Mode 1 and synchronized to the grid.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

7/19/16	-2-PAT-7.0, Test Sequence for 75% Plateau performance section initiated. -2-PAT-1.2, Load Swing Test, was completed with all Acceptance Criteria met. CR 1193637 was written for the Review Criteria not being met for an undershoot of steam header pressure. The Review Criteria required an undershoot of no more than 25 psi and the actual was 28.5 psi. This test was originally scheduled for the previous 50% Plateau testing, however, issues with the turbine IMP IN controls prevented performance during that plateau. The unit was held at 45% power on the ascension to the 75% testing plateau to perform this test. -2-PAT-1.4 Pipe Vibration Monitoring, applicable sections for the load swing were completed with all criteria met.
7/25/16	-Unit 2 reached 75% Plateau testing power level. -2-PAT-1.5, Loose Parts Monitoring System, was completed with all criteria met. CR 1171424 documents three channels removed from service. -2-PAT-1.10, Integrated Computer System (ICS), completed. CR 1195476 written for failure of Acceptance Criteria. MCR indicator 2-TI-62-71 comparison to ICS PID T0127A (Regen Heat Exch Letdown Temp) was not within the MED. The CR was closed after calibration of the instrument. -2-PAT-1.11, RVLIS Performance Test, applicable sections completed with all criteria met. -2-PAT-1.12, Common Q Post Accident Monitoring System, applicable sections were completed with all criteria met. -2-PAT-1.8, Thermal Expansion of Piping Systems, completed with all criteria met.
7/26/16	-2-PAT-1.4, Pipe Vibration Monitoring, completed with CR 1195665 written on excessive vibration on the Main Steam Line Trap drain line. Temporary repairs to stabilize the line were initiated. All other criteria were met.
7/27/16	-2-PAT-3.3, RCS Flow Measurement completed with all criteria met.
7/28/16	-2-PAT-1.6, Startup Adjustments of Reactor Control System, was completed with all criteria met. -2-PAT-7.1, Calibration of Steam and Feedwater Flow Instruments at 75% Power, was completed. Steam Flow and Feedwater Flow data obtained in Section 6.1 of this PAT on 7/26/16 was used to adjust the span of the associated Steam Flow transmitters. Post calibration data was subsequently taken in accordance with Section 6.2 of this PAT on 7/28/16. and all Review Criteria were met. There was no Acceptance Criteria for this PAT.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

7/29/16	-2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, was completed with all Acceptance Criteria met and all Review Criteria met upon the second performance. CR 1196243 and CR 1196245 were generated for the initial failures. On the second data collection all Review Criteria were met and the CRs closed. -2-PAT-1.9, Automatic Steam Generator Level Control, was completed with all criteria met. -2-PAT-7.0, Test Sequence for 75% Plateau, was TRG approved.
8/1/16	-2-PAT-8.0, Test Sequence for 100% Plateau, performance section was initiated. Due to increasing generator bushing temperatures and concerns for further power increase, the original sequence of testing was revised to perform 2-PAT-8.5, Shutdown From Outside the Main Control Room. An outage after the PAT performance was planned for repairs to the generator bushing.
8/3/16	-Unit 2 power reduced to approximately 30% RTP. -2-PAT-8.5, Shutdown From Outside The Main Control Room was completed with all criteria met. The Unit was held in Mode 3 for equipment repairs.
8/7/16	-Repairs were completed and Unit 2 startup initiated. -Unit 2 entered Mode 2 at 12:22 and the reactor was critical at 12:31. -Unit entered Mode 1 at 16:14.
8/9/16	-Unit 2 synchronized to the grid at 06:12. A delay in synchronization occurred due to particles in the thrust bearing wear trip fluid which required flushing multiple times.
8/10/16	-During power ascension an issue with increased temperatures on C phase main generator bushing developed. This temperature issue was noted prior to the Shutdown from Outside the Main Control Room and resulted in a second planned outage.
8/13/16	-Unit 2 was manually tripped at 03:06 and stabilized in Mode 3 for a planned outage.
8/22/16	-Unit 2 reactor critical at 02:28. -Mode 1 entry at 08:32. -Unit 2 synchronized to the grid at 13:53.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

8/23/16	-Unit 2 reactor was manually tripped at 13:56 when the 2A Main Turbine Driven Feedwater Pump slowed and failed to provide sufficient flow to maintain steam generator levels. The unit was stabilized in Mode 3.
8/25/16	-Unit 2 entered Mode 2 at 14:27 and the reactor was critical at 14:46. -Unit 2 entered Mode 1 at approximately 17:25. -Unit 2 synchronized to the grid at 23:19.
8/29/16	-Unit 2 at 93% rated thermal power allowing PAT testing to commence at the 90% plateau. -2-PAT-1.6, Startup Adjustments of Reactor Control System, completed. For the 90% Plateau data collection was completed and satisfactory for this plateau. CR 1208178 was initiated before performance because 2-PT-1-81 was unavailable for the test due to a steam leak. Results were acceptable for continuation to the 100% plateau where measurements were repeated. -2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, Review Criteria 5.2.B and 5.2.C were not met but a CR was not written as the 2-PAT-1.7 performance was designed to correct the issue and the Acceptance Criteria were verified at the 100% power plateau. All other Review and Acceptance Criteria were met. -2-PAT-8.4, Calibration Of Steam And Feedwater Flow Instruments at 100% Power, performance at 93% completed. All Review Criteria were met for Section 6.1. There was no Acceptance Criteria for Section 6.1.
8/30/16	-Unit 2 at > 98% rated thermal power allowing PAT testing to commence at the 100% plateau. -2-PAT-1.5, Loose Parts Monitoring System was completed with all criteria met. CR 1171424 documents three channels removed from service. -2-PAT-1.6, Startup Adjustments of Reactor Control System, data collection was field work complete before the unit tripped. Subsequently, CR 1211020 was written for one failed Acceptance Criteria. The failed criteria was due to full load steam pressure being below the expected value because T_{avg} was at its maximum value. However, there is no safety or operational concern. Additionally, CR 1211015 was written for failed Review Criteria . 2-PT-1-81 was out of service, therefore, calibrations of the pressure transmitter will be verified when 2-PT-1-81 is returned to service outside the PAT program. CR 1208178 was previously written for this issue and WO 118121693 will resolve the issue.

3.0 **Watts Bar Unit 2 Startup Chronology** (continued)

8/30/16	-2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, data collection was field work complete before unit tripped. Data reduction was completed with failure to meet Acceptance Criteria for Loop 4 T_{avg}. On 9/7/16 CR 1211021 was written documenting this failed Acceptance Criteria. Although a failure, there was no safety concern or failure to meet the licensing basis. All Review Criteria were met on this performance of the PAT. -2-PAT-1.8, Thermal Expansion of Piping Systems, completed with all criteria met. -2-PAT-1.10, Integrated Computer System (ICS), was completed for the 100% Plateau. CR 1208754 was generated for failure of meeting the MED between indicator 2-TI-062-0004 and ICS point T0181A, RCP 1 No 1 Seal Outlet Temperature. A WO was generated to calibrate and has subsequently closed. -2-PAT-1.11, RVLIS Performance Test, was completed with all criteria met. -2-PAT-1.12, Common Q Post Accident Monitoring System, was completed with all criteria met. -2-PAT-3.3, RCS Flow Measurement, data collection was field work complete before unit tripped. -2-PAT-8.4, Calibration of Steam and Feedwater Flow Instruments, at 100% Power, was field work complete. CR 1208875 was written to document failure of Review Criteria on three steam flow transmitters. WOs were initiated to respan the transmitters. -Unit 2 received an automatic Turbine Trip - Reactor Trip at 21:09:13 due to a fault in the 2B Main Bank Transformer, resulting in a fire in the transformer. The unit was stabilized and subsequently placed in Mode 4 for repairs.
9/15/16	-2-PAT-8.6, Plant Trip from 100% Power, was evaluated from data gathered during the actual plant trip on 8/30/16. All Acceptance Criteria was met. CR 1209770 was written to evaluate the equivalency of the data collected by the plant as well as one Review Criteria which did not meet pressurizer level modulation to no load setpoint within 30 minutes. A Westinghouse evaluation concluded the response was acceptable. -2-PAT-1.4, Pipe Vibration Monitoring, Section 6.6.19 was closed based on an engineering walkdown evaluation CR 1211196.
9/25/16	-Unit 2 entered Mode 1 at 01:53 after the Spare Main Bank Transformer was placed in service for the failed 2B Main Bank Transformer which was removed from site.
9/26/16	-Unit 2 generator synchronized to the grid at 01:07.

3.0 Watts Bar Unit 2 Startup Chronology (continued)

9/27/16	-Unit 2 reached 100% power. -RCI-159, Radiation Baseline Surveys completed for 100% Plateau. No Acceptance or Review Criteria were associated with this procedure.
9/28/16	2-PAT-1.9, Automatic Steam Generator Level Control, was field work complete with all criteria met.
9/29/16	-2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, was completed with failure to meet Acceptance Criteria for Loop 4 T_{avg}. However, on 9/7/16 CR 1211021 had been previously written documenting this failed Acceptance Criteria.
9/29/16	-2-PAT-1.2, Load Swing Test, was field work complete with all Acceptance Criteria met. CR 1218746 was written for failure of one Review Criteria for S/G Level response. Westinghouse evaluated the response to be adequate with no further testing required. -2-PAT-3.3, RCS Flow Measurement, was completed with all criteria met. -2-PAT-8.4, Calibration of Steam and Feedwater Flow Instruments, at 100% Power, was field work complete with all criteria met.
9/30/16	2-PAT-1.3, Large Load Reduction Test, was field work complete with all Acceptance Criteria met. CR 1218917 was written for Review Criteria failure of S/G levels to remain within ±15% of the program level. Westinghouse concluded the response was acceptable. 2-PAT-1.4, Pipe Vibration Monitoring, was completed with CR 1208694 written for main steam traps excessive vibration as was noted at the 75% Plateau also. Civil Design generated WO 118122821 to design and install a restraint outside the PATP.
10/3/16	2-PAT-1.6, Startup Adjustments of Reactor Control System, was field work complete on 10/3/16. Previously, CR 1211020 was written for one failed Acceptance Criteria. The failed criteria was due to full load steam pressure being below the expected value because T_{avg} was at its maximum value. However, there is no safety or operational concern. Additionally, CR 1211015 was written for failed Review Criteria. Due to 2-PT-1-81 being out of service calibrations of the pressure transmitter will be verified when 2-PT-1-81 is returned to service outside the PAT program. CRs 1208178 and 1216904 were previously written for this issue.
10/6/16	-2-PAT-8.0, Test Sequence for 100% Plateau was TRG approved.

4.0 INITIAL FUEL LOAD

4.1 Overview and Summary of Initial Core Loading

The initial core loading at WBN Unit 2 was accomplished in approximately 76 hours from December 4, 2015, to December 8, 2015, as directed by 2-PAT-2.0, Initial Core Loading Sequence.

Core loading was performed "wet" with the refueling cavity and the reactor vessel filled with refueling concentration borated water at normal refueling levels. The core loading sequence was performed in accordance with an approved Fuel Assembly Transfer Form (FATF). Actual movement of fuel was performed in accordance with 2-FHI-7, Fuel Handling and Movement, as directed by 2-PET-105, Initial Core Loading.

The neutron monitoring station for Inverse Count Rate Ratio determinations were established in the main control room to monitor source range detectors N-31 and N-32. ICRR plots were maintained for these detectors during all core loading sequence steps and during delays in core loading to ensure that an adequate subcritical margin was maintained at all times.

As a visual aid in tracking fuel movement evolutions and to ensure the core load configuration was in accordance with the approved loading pattern prescribed on the FATF, a core status display was maintained in the Main Control Room.

RCS boron concentration was monitored during core loading to ensure that the boron concentration remained within prescribed limits.

Some fuel assemblies were required by plan to be moved more than once, specifically those bearing primary neutron sources. As such, the core loading sequence required two in-core fuel assembly movements to move the source bearing fuel assemblies from the reactor baffle wall to their final locations within the core. This was done to ensure that neutron counts could be monitored by the Source Range instrumentation at all times during core loading. After the core was loaded, a video recording was made and verification of proper fuel assembly position and orientation was conducted. Fuel Related Components (FRCs) were confirmed to be inserted into the proper fuel assembly with the proper orientation in the Spent Fuel Pool prior to core load. The final core load configuration was consistent with the Westinghouse Core Loading Plan for Unit 2 Cycle 1.

4.2 Initial Core Loading Sequence (2-PAT-2.0)

This test started on 11/19/15 with prerequisites and completed on 12/8/15.

1.0 Test Objectives

The objectives of this test were to:

1.1 Sequence the procedures that established the prerequisites required for the initial core loading of Unit 2

1.2 Define the sequence of operations and tests which were to be conducted during and following completion of the initial core loading.

The following PATs/PETs/RCI were sequenced for performance by 2-PAT-2.0:

- 2-PAT-2.1 Reactor System Sampling for Core Load
- 2-PAT-2.2 Response Check of Core Load Instrumentation After 8 Hour Delay in Fuel Movement
- 2-PET-102 Pre-Power Escalation NIS Calibration Data
- 2-PET-105 Initial Core Loading
- RCI-159 * Radiation Baseline Surveys

Note: * Indicates that the test is performed at multiple test plateaus. The description of the testing is documented in the section (plateau) in which it was completed.

2.0 Test Methods

Pre-requisite actions started on 11/19/15, prior to entry into Mode 6 to establish prerequisite conditions in support of commencement of initial core loading. The test continued through verification of core loading and was field complete on 12/8/15, prior to the reassembly of the reactor vessel in preparation for Mode 5 entry.

The major pre-requisites included the following:

- Verification all Preoperational Test completed and test results approved or technical justifications for delaying tests until after fuel load were approved by the Plant Manager
- Verification 2-PET-102, Pre-Power Escalation NIS Calibration Data, was successfully completed to the extent necessary
- RCI-159 Radiation Baseline Surveys commenced for the pre-fuel load survey

4.2 Initial Core Loading Sequence (2-PAT-2.0) (continued)

- 2-PAT-2.1, Reactor System Sampling for Core Load, started
- Visual Inspection of the reactor vessel core support plate in accordance with 2-PET-105, Initial Core Loading completed Testing included the following:
- 2-PAT-2.1, Reactor System Sampling for Core Load, completed on 12/4/15 with all criteria met.
- 2-PAT-2.2, Response Check of Core Load Instrumentation After 8 Hour Delay in Fuel Movement, completed on 12/5/15 with all criteria met
- 2-PET-102, Pre-Power Escalation NIS Calibration Data, applicable sections completed with all criteria met
- 2-PET-105, Initial Core Loading completed on 12/8/15 with all criteria met.
- RCI-159, Radiation Baseline Surveys, completed on 12/3/15. There was no Acceptance or Review Criteria associated with this procedure.

3.0 <u>Test Results</u>

All Acceptance/Review Criteria were contained within the tests sequenced by this test.

4.0 <u>Problems</u>

Problems encountered are addressed in the following discussions of each test sequenced by 2-PAT-2.0.

4.3 Reactor System Sampling for Core Load (2-PAT-2.1)

This test was performed as part of test sequence 2-PAT-2.0, Initial Core Loading. Testing was started on 11/19/15 and field work completed on 12/4/15.

1.0 Test Objectives

The objectives of this test were to:

1.1 Verify the boron concentrations in the Reactor Coolant System (RCS), Residual Heat Removal (RHR) system and other directly connected portions of auxiliary systems are uniformly borated to prevent inadvertent dilution during core loading.

1.2 Verify un-borated water sources are configured to prevent inadvertent dilution during core loading.

1.3 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 4, Reactor System Sampling For Core Loading Test Summary.

2.0 Test Methods

The preliminary actions of 2-PAT-2.1 researched logs and procedurally driven Unit 2 activities that were completed and that were associated with the preparations of the unit 2 water systems for entering Mode 6. The research started with ensuring the RWST was borated to (3100 to 3300) ppm. Actual recorded samples of the RWST were 3326 ppm on 09/07/2015 and 3224 ppm on 11/21/2015. This confirms a correctly borated RWST. This RWST water was subsequently used to fill the RCS and partially fill the Refueling Canal and Cavity. Later boron sample results showed Refueling Canal at 3290 ppm and Refueling Cavity at 3281 ppm.

Following the proper boration of the RWST and water transfer to the RCS, unit activities were verified that circulated water through:

Both RHR A-A and B-B pump mini flows
Both RHR pumps
RHR to CVCS Letdown
Both Charging pumps
Both Containment Spray pumps (re-circulated borated RWST)
Refueling Water Purification Pump B

Refueling Water Purification Pump A was found to be tagged with a Caution Order 0-CO-2015-0048 stating that the pump has high vibration. WO 116318322 was previously written to address this issue. The volume of potentially diluted water was conservatively calculated to be 13 gals. This small volume did not pose any risk to challenging any criteria listed in this PAT.

4.3 Reactor System Sampling for Core Load (2-PAT-2.1) (continued)

The Boron Injection Tank (BIT) was verified to have been borated and mixed via the performance of 2-SI-63-905, Boron Injection Check Valve Flow During Refueling Outages.

It was verified that both trains of Safety Injection were circulated during the performance of 2-SI-63-906, Safety Injection Check Valve Full Flow Testing During Refueling Outages, on 11/25/2015.

All 4 Cold Leg Accumulators were verified by sample to be correctly borated with the lowest reading 3175 ppm and the highest reading 3206 ppm.

The water in the Holdup Tank B (HUT B) was recirculated and sampled for boron concentration on 11/04/15 and found to be 3204 ppm boron; this water was used to fill the Fuel Transfer Canal. The Spent Fuel Pool (SFP) was sampled for boron concentration on 11/23/15 and found to be 3261 ppm boron. Boric Acid Tanks B and C were sampled for boron concentration on 11/21/15 and found to be 6919 and 6808 ppm boron respectively.

Problems encountered while running 2-SI-63-905 and 2-SI-63-906 resulted in a partial drain down moving water back to the RWST. This same water was again used to fill the Refueling Cavity and Chemistry re-performed 2-SI-78-1, Reactor Coolant System and Refueling Canal Refueling Operations Boron Determination, to document compliance with the refueling boron concentration requirements.

Watts Bar Unit 2 systems connected to the Reactor Coolant System (RCS) were adequately borated and mixed to prevent a dilution event in support of the initial core loading operations.

2-PAT-2.1, Reactor System Sampling For Core Load, supported this conclusion from research of the chemistry and operation logs. Configuration control measures were in place to ensure that the RCS and connected systems remain adequately borated for support of the initial core loading operations. The Unit 2 Refueling Water Storage Tank (RWST) was at approximately 16.9% and based on calculations of recent makeup to the RWST it was determined that the tank was adequately borated. Technical Specifications required Operations to validate compliance with the RWST boron concentration and level prior to Mode 6. Technical Specifications required Watts Bar Unit 2 to maintain Mode 6 surveillance instructions in frequency. Therefore, no additional sampling or mixing was required for 2-PAT-2.1.

4.3 Reactor System Sampling for Core Load (2-PAT-2.1) (continued)

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 Boron Concentration of samples meet requirements of the Technical Specifications.

3.1.1 The RCS Boron concentration is greater than or equal to 3100 ppm and less than or equal to 3300 ppm.

The RCS Boron as measured in the RHR TRAIN B system was 3284 ppm.

3.1.2 The boron concentration final samples obtained from the designated sample points identified are uniformly borated between 3100 ppm and 3300 ppm.

The boron concentration for sample points met the requirements.

3.1.3 The boron concentration of samples obtained from the Boric Acid Tanks (BAT B and BAT C) are within the limits of 6120 $\leq C_B \leq$ 6990 ppm.

Boron concentrations were 6919 ppm in BAT B and 6808 ppm in BAT C.

3.1.4 Un-borated water sources are configured to prevent inadvertent dilution during core loading.

2-SI-62-1, Uncontrolled Boron Dilution Paths, was satisfactorily completed for Mode 6.

Review Criteria

3.2 The boron concentrations for the Reactor Coolant System (RCS) and directly connected portions of the auxiliary systems are greater than or equal to 3100 ppm and less than or equal to 3300 ppm.

Boron concentrations for the RCS and directly connected portions of the auxiliary systems met the requirement.

4.0 Problems

There were no significant problems encountered during the performance of this test.

4.4 Response Check of Core Load Instrumentation After 8 Hour Delay in Fuel Movement (2-PAT-2.2)

This test was performed as part of test sequence 2-PAT-2.0, Initial Core Loading Sequence. Testing was started and completed on 12/5/15.

1.0 Test Objectives

The objective of this test was to:

1.1 Verify response of the Source Range Channels prior to resumption of fuel loading following a delay of eight (8) hours or more.

2.0 Test Methods

Three methods of testing were available for use:

2.1 Statistical Evaluation Method using the Scaler Timer

2.2 Statistical Evaluation Method using the Source Range Count Rate indications.

2.3 Response Check of Core Load Instrumentation Using Primary Source Bearing Fuel Assembly Movement.

The Statistical Evaluation Method using the Scaler Timer provided the verification of the Acceptance Criteria for resumption of fuel movement.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

<u>Acceptance Criteria</u>

3.1 The Source Range instrumentation for both channel N-31 and N-32 were evaluated and determined to be acceptable for continuation of fuel loading by meeting at least one Review Criteria.

Review Criteria 3.2, below

4.4 Response Check of Core Load Instrumentation After 8 Hour Delay in Fuel Movement (2-PAT-2.2) (continued)

Review Criteria

3.2 Statistical Evaluation Method using the Scaler Timer:
Statistical Reliability Factor (SRF) for Source Range Channels shall be ≥ 0.5 and ≤ 1.4.

Results indicated the SRF for Source Range Channel N-31 was 1.2395 and SRF for Source Range Channel N-32 was 0.8609.

3.3 Statistical Evaluation Method using the Source Range Count Rate indications:
1. The Student F Distribution Test shall be satisfied by having $F_{exp} \leq 3.179$.
2. The Student T Distribution Test shall be satisfied by having $T_{exp} \leq 2.101$.

This method was originally chosen, however, problems were encountered. See Problems below.

3.4 Neutron instrumentation (Source Range Channels N-31 and N-32) are operational and indicates a positive (negative) change in count rate as the neutron level detected from a source is increased (decreased).

This method was not used.

4.0 <u>Problems</u>

[1] No CR initiated:
Section 6.2, Statistical Evaluation Using Source Range Count Rate Indications, method was attempted four (4) times with unsuccessful results. Based on only three assemblies loaded at the time of performance, low counts appeared to cause data scatter which was observed in monitored count rates. This failure of Section 6.2 method resulted in the transition to Section 6.1, Statistical Evaluation Method using the Scaler-Timer. Section 6.1 method was acceptable. No CR was initiated since this was a potential scenario and the test provided alternative methods.

4.5 Pre-Power Escalation NIS Calibration Data (2-PET-102)

This test was performed as part of the test sequence 2-PAT-2.0, Initial Core Loading Sequence. The performance of 2-PET-102 was conducted via WO 116884907. The WO started 08/25/2015 and was complete on 04/12/16 with calibration of all Power Range and Intermediate Range detectors.

1.0 <u>Test Objectives</u>

The objectives of this test were to:

1.1 Provide Nuclear Instrumentation System (NIS) Power Range (PR) and Intermediate Range (IR) excore detector calibration data.

1.2 Initiate an adjustment of the NIS before startup for a new fuel cycle.

Note: The calculation methodology in 2-PET-102 applied to the changes expected to occur due to a refueling outage. This procedure accommodated the calibrations to be performed for Cycle 1.

2.0 <u>Test Methods</u>

The normal method for determining calibration data after fuel reload and prior to startup is to ratio the sum of selected weighted assembly predicted powers from the Beginning of Life (BOL) of the previous fuel cycle (Unit 1 Cycle 1 was used as the reference condition) to the BOL of the upcoming cycle. This ensures a ratio based upon similar BOL core conditions including the neutron energy spectrum and a nearly cosine axial flux shape. This provides the most accurate excore Axial Offset indications for the power range channels. This same methodology results in the most accurate power indications for the intermediate range channels.

This same methodology was used to predict the Intermediate Range and Power Range calibration setpoints for the Unit 2 Cycle1 startup, except that Unit 1 Cycle 1 is used as the reference condition. In this case, average composite values for the channels were used.

3.0 <u>Test Results</u>

All Acceptance/Review Criteria were met or resolved as delineated below.

4.5 Pre-Power Escalation NIS Calibration Data (2-PET-102) (continued)

Acceptance Criteria

3.1 Power Range Channel calibrations have been completed.

The Power Range channel calibrations were completed via:

- WO 115898162
- WO 115898208
- WO 115898252
- WO 115899187

3.2 Intermediate Range Channel predicted full power adjustments have been completed.

Intermediate Range Channel predicted full power adjustments were performed. IR Gain Adjustment potentiometers were set to the values calculated in the PET via steps in Section 7.0 of the PET.

3.3 Intermediate Range Channel Operational Tests (COTs) have been completed.

The Intermediate Range channel COTs were completed via:

- WO 117499181
- WO 117499184

Review Criteria

None

4.0 **Problems**

There were no significant problems encountered during the performance of this test.

4.6 Initial Core Loading (2-PET-105)

This test was performed as part of test sequence 2-PAT-2.0, Initial Core Loading Sequence. 2-PET-105 testing via WO 117370408 started on 11/23/2015 with the verification of Unit 2 fuel assemblies and component inserts in the Spent Fuel Pool and completed on 12/08/2015 with the completion of fuel load and core load verification.

1.0 Test Objectives

The objectives of this test were to:

1.1 Identify the activities and requirements for fuel loading which ensure fuel loading is conducted in a cautious and controlled manner:

1.1.1 Specify the sequence for loading fuel assemblies into the reactor vessel such that the final core configuration is consistent with that specified in the NuPOP for current fuel cycle. See Figure 4.6-1, U2C1 Core Load Sequence.

1.1.2 Specify the fuel assembly identification number and type of insert for each core location.

1.1.3 Establish the requirements for periodic and continuous neutron monitoring during each step of the core loading process.

1.1.4 Prescribe the steps necessary for obtaining and evaluating neutron monitoring data during core loading.

1.1.5 Identify the neutron monitoring channels to be used during each step of the core loading sequence to ensure subcritical conditions are maintained.

1.2 Satisfied the requirements of UFSAR Table 14.2-2, Sheet 3, Initial Fuel Loading Test Summary.

2.0 Test Methods

Only data from "responding" detectors identified by the data package was used in evaluating the safety of continued core loading. Prior to completing the loading of the initial nucleus of eight fuel assemblies, significant changes in the ICRR data were expected to occur due to geometry effects arising from changes in detector-to-fuel assembly coupling. Therefore, the ICRR values were re-normalized following movement of source bearing fuel assemblies from the baffle wall to their final location(s) in the core.

Changes in neutron flux level during and following fuel assembly insertion was monitored for indications of abnormal and/or unstable reactivity behavior.

All fuel movement was performed in accordance with 2-FHI-7, Fuel Handling and Movement.

The core status display in the main control room was updated, as required, to reflect the actual physical location of all fuel assemblies and fuel related components at all times during the core loading evolution.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Note: Unit 2 Core Load ICRR plot is provided in Figure 4.6-2.

Acceptance Criteria

3.1 The core was successfully loaded in accordance with the Unit 2 Cycle 1 Westinghouse Core Load Plan.

Verification of successful core loading was provided via 2-TI-28, Physical Verification of Core Load Prior to Vessel Closure, (WO 117370592.) See Figure 4.6-1.

3.2 At completion of each mini-core, the final count rate from any detector shall not unexpectedly double from the initial count rate before the assembly was inserted.

Neutron count rates observed during fuel movement did not unexpectedly double at any time.

3.3 At completion of each mini-core the ICRR response from any detector shall not be less than 0.5 as each fuel assembly is inserted.

Neutron count rates observed during fuel movement did not unexpectedly double at any time and ICRR remained above 0.5, see Figure 4.6-2.

3.4 Core loading operations are required to be immediately stopped and the Containment Building evacuated if any of the following conditions occur during core loading. Movements of an active source bearing assembly, or detector-to-fuel assembly neutronic coupling are anticipated type changes.

4.6 **Initial Core Loading (2-PET-105)** (continued)

3.4.1 An unanticipated simultaneous increase in the neutron count rate by a factor of > 2 on all "responding" neutron monitoring channels.

3.4.2 An unanticipated simultaneous increase in the neutron count rate on any individual "responding neutron monitoring channel by a factor of $\geq$ 5.

Neutron count rates were acceptable and did not meet either criteria to warrant suspension of core loading operations or evacuation of the Containment Building.

<u>Review Criteria</u>

All required Review Criteria for this test were met as delineated below:

3.5 Assessment of the ICRR response should be based on the predicted ICRR response.

ICRR plots maintained during core loading activities contained both actual plant ICRR data as well as predicted ICRR data from the fuel vendor, see Figure 4.6-2.

3.6 Placement of initial fuel assemblies up to placement of primary source assemblies in final core location should be detected by the ICRR response.

ICRR monitoring was maintained at all times during core loading, including loading of the first "mini-cores" and final movement of primary source bearing fuel assemblies.

3.7 ICRR response should not be less than 0.8 for any fuel assembly after the primary source assemblies have been placed in their final locations.

ICRR data during core loading was determined to be less than 0.8 following final placement of the two source bearing assemblies. CR 1112886 was initiated to document violation of this Review Criterion. Violation of this criterion does not represent a failure of this test, as it only requires further evaluation by the fuel vendor. The fuel vendor was notified and agreed that the data was acceptable.

4.6 Initial Core Loading (2-PET-105) (continued)

4.0 Problems

[1] CR 1112049 was written to document a labeling issue. Source Range labeling difference was noted between 2-PET-105 and the Unit 2 Main Control Room. Urgent Change 1 was processed for 2-PET-105 to correct the labeling issue.

[2] CR 1112886 was initiated to document violation of the 0.8 ICRR limit during core loading, as described in Section 3.7.

[3] CR 1112204 was initiated to document foreign material, later determined to be glue, on the bottom nozzles on multiple fuel assemblies during initial core load. All assemblies noted to have debris were cleaned prior to being loaded in the reactor. Efforts to remove the debris caused schedule delays.

4.6 Initial Core Loading (2-PET-105) (continued)

FIGURE 4.6-1
U2C1 Core Load Sequence

U2C1 Core Load Sequence

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	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	
15	↑ N				N28 61 PD316	N35 62 8W0077	N14 5 PD305	N43 4 12W0064	N25 3 PD296	N40 2/20 8W0073	N11 19 PD318					15
14			N04 65 PD312	N30 66 R202	N51 64 16W0047	L55 63 R205	N64 7A 23PS4	L05 7 R199	N57 6 24W0007	L18 21 R194	N56 22 16W0042	N06 24 R174	N26 23 PD320			14
13		N17 70 PD284	N48 71 12W0061	M12 69 20W0028	L32 68 R179	M24 67 20W0046	L40 18 R214	M44 9 20W0022	L44 8 R203	M26 25 20W0026	L23 26 R207	M35 27 20W0054	N41 29 12W0063	N20 28 PD301		13
12		N32 72 R222	M11 73 20W0047	L24 103 R212	M21 104 20W0029	L65 105 PD304	M22 106 20W0025	L63 117 R181	M14 113 20W0067	L66 112 PD297	M06 111 20W0063	L25 110 R200	M13 31 20W0059	N21 30 R201		12
11	N27 74 PD298	N49 75 16W0044	L02 76 R220	M05 107 20W0051	L04 108 R182	M46 109 20W0062	L20 118 PD285	M04 119 20W0037	L62 120 PD302	M47 116 20W0031	L48 115 R188	M42 114 20W0056	L19 34 R221	N55 33 16W0046	N29 32 PD290	11
10	N33 77 8W0075	L09 78 R216	M40 121 20W0065	L10 122 PD292	M23 123 20W0032	L27 124 R171	M55 125 24W0004	L17 131 R183	M54 130 24W0011	L37 129 R211	M37 128 20W0027	L61 127 PD291	M18 126 20W0041	L39 36 R192	N39 35 8W0079	10
9	N07 79 PD299	N62 101 24W0005	L51 174 R186	M07 177 20W0043	L16 180 PD303	M57 183 24W0014	L34 189 PD282	M56 190 24W0017	L35 188 SS6	M51 171 24W0013	L22 168 PD313	M43 165 20W0052	L11 162 R206	N61 59 24W0010	N08 37 PD308	9
8	N47 99 12W0057	L21 102 R210	M09 175 20W0035	L08 178 R197	M38 181 20W0048	L67 184 R173	M60 192 24W0018	L07 193 R193	M49 191 24W0009	L68 172 R198	M28 169 20W0055	L59 166 R175	M31 163 20W0068	L57 60 R176	N46 57 12W0060	8
7	N22 98 PD317	N60 100 24W0001	L36 173 R178	M48 176 20W0050	L53 179 PD310	M59 182 24W0016	L30 186 SS5	M58 187 24W0006	L12 185 PD306	M53 170 24W0012	L60 167 PD300	M32 164 20W0061	L14 161 R217	N59 58 24W0003	N01 56 PD311	7
6	N38 96 8W0078	L54 97 R191	M39 155 20W0042	L42 156 PD294	M41 157 20W0038	L46 158 R177	M52 159 24W0008	L69 160 R204	M50 154 24W0002	L47 153 R225	M10 152 20W0040	L56 151 PD314	M15 150 20W0044	L58 55 R184	N37 54 8W0074	6
5	N12 93 PD295	N54 94 16W0041	L15 95 R213	M33 143 20W0045	L26 144 R195	M01 145 20W0021	L49 147 PD322	M19 148 20W0036	L01 149 PD288	M16 142 20W0030	L31 141 R196	M27 140 20W0049	L28 53 R185	N53 52 16W0048	N09 51 PD321	5
4		N16 91 R218	M25 92 20W0053	L41 136 R215	M30 137 20W0064	L38 138 PD283	M29 139 20W0058	L45 146 R172	M08 135 20W0033	L43 134 PD307	M45 133 20W0057	L13 132 R219	M02 50 20W0060	N15 49 R209		4
3		N19 89 PD319	N45 90 12W0058	M36X 88 20W0039	L64 87 R170	M34 86 20W0023	L03 17 R224	M20 16 20W0024	L52 15 R223	M17 44 20W0034	L50 45 R180	M03 46 20W0066	N44 48 12W0062	N18 47 PD287		3
2			N05 84 PD315	N31 85 R208	N52 83 16W0045	L33 82 R189	N63 14A 23PS3	L29 14 R226	N58 13 24W0015	L06 40 R190	N50 41 16W0043	N02 43 R187	N24 42 PD286			2
1	L 2.1% ENR.	M 2.6% ENR.			N23 80 PD289	N36 81 8W0080	N13 12 PD281	N42 11 12W0059	N10 10 PD293	N34 1/39 8W0076	N03 38 PD309				N 3.1% ENR.	1
	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	

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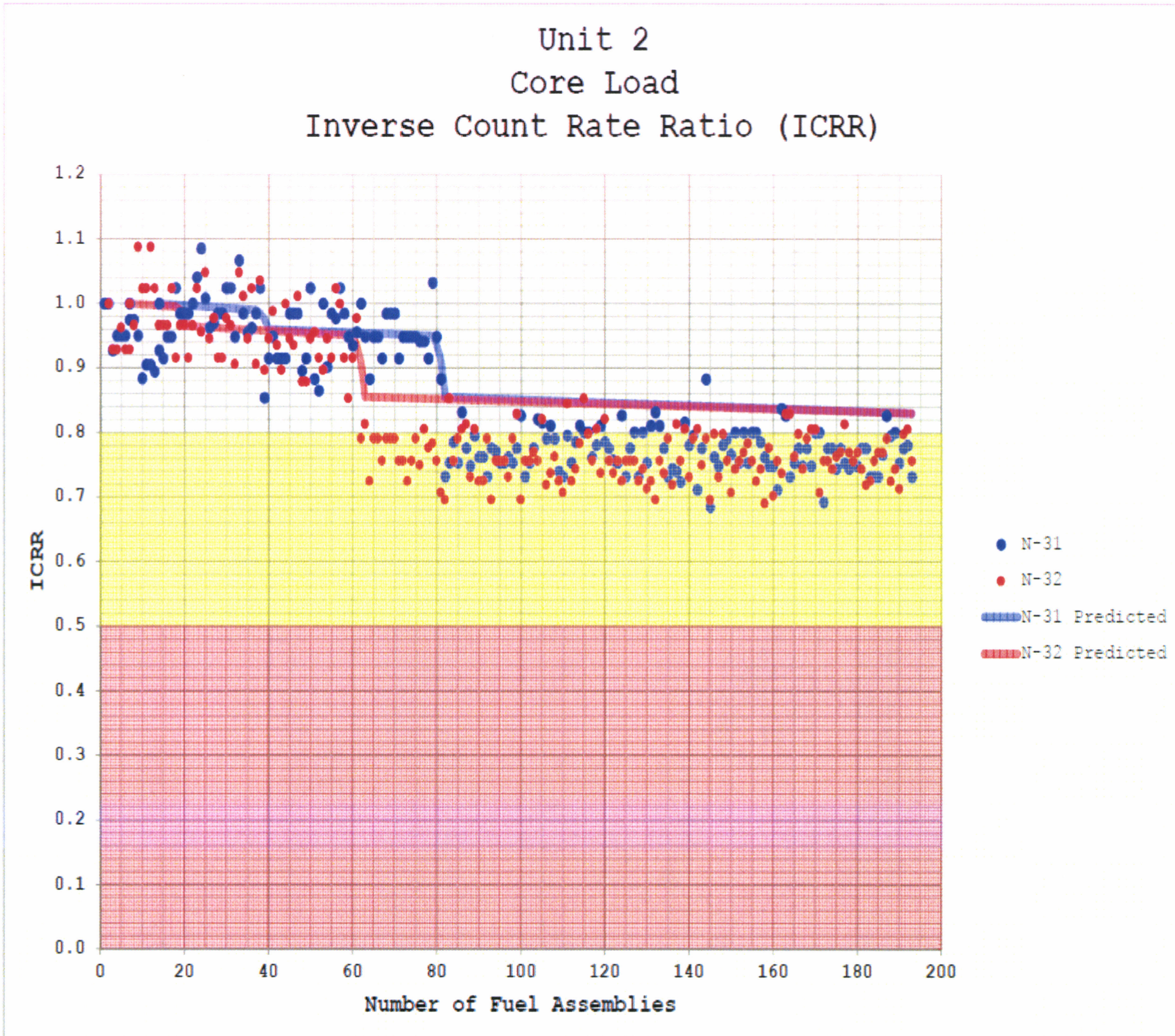
9/11/15
PERFORMED BY / DATE

9/11/15
VERIFIED BY / DATE

4.6 Initial Core Loading (2-PET-105) (continued)

FIGURE 4.6-2

Unit 2 Core Load ICRR



5.0 PRECRITICAL TESTING

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0)

This test started on 12/9/2015 and was completed on 05/15/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Serve as controlling document for establishing the required pre-requisite conditions to permit testing following the completion of 2-PAT-2.0.

1.2 Govern the sequence of tests performed in Mode 6 through Mode 3.

1.3 Implement testing deferred from Pre-Operational Test Instruction, 2-PTI-062-03, HFT Charging and Letdown documented in CR 1075347 and CR 1085430.

The following PATs/PETs/RCI were sequenced for performance by 2-PAT-3.0:

- 2-PAT-1.4 * Pipe Vibration Monitoring
- 2-PAT-1.6 * Startup Adjustments of Reactor Control System
- 2-PAT-1.7 * Operational Alignment of Process Temperature Instrumentation
- 2-PAT-1.8 * Thermal Expansion of Piping Systems
- 2-PAT-1.11 * RVLIS Performance Test
- 2-PAT-1.12 * Common Q Post Accident Monitoring System
- 2-PAT-3.1 Control Rod Drive Mechanism Timing and CERPI Initial Calibration
- 2-PAT-3.2 Pressurizer Spray Capability and Continuous Spray Flow Setting
- 2-PAT-3.3 * RCS Flow Measurement
- 2-PAT-3.4 Rod Control and Rod Position Indication (CERPI)
- 2-PAT-3.7 Reactor Coolant Flow Coastdown
- 2-PAT-3.8 Rod Drop Time Measurement and Stationary Gripper Release Timing
- 2-PAT-3.10 Reactor Trip System
- 2-PAT-3.11 Adjustment of Steam Flow Transmitters at Minimal Flow
- 2-PAT-5.1 * Dynamic Automatic Steam Dump Control
- 2-PET-106 Control Rod Drive Mechanism Timing
- RCI-159 * Radiation Baseline Surveys

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0) (continued)

Note: * Indicates that the test is performed at multiple test plateaus. The description of the testing is documented in the section (plateau) in which it was completed.

2.0 Test Methods

Prerequisite actions for this Power Ascension Test (PAT) started on 12/9/2015 and completed on 12/11/2015 and included verification of the following major items:

- 2-PAT-2.0 Initial Core Loading Sequence completed.
- 2-GO-7 Refueling Operations performed concurrently with 2-PAT-3.0.
- 2-GO-10 Reactor Coolant System Drain and Fill Operation performed concurrently with 2-PAT-3.0.
- RCI-159 Radiation Baseline Surveys, commenced for post-fuel load activities.

Testing was performed at eight defined plateaus including, ambient (<105°F), 250°F, 300°F, 360°F, 400°F, 450°F, 500°F, and 557°F. This report is a summary therefore see individual test packages for specific details at each plateau.

Ambient Plateau testing included the following:

- RCI-159, Radiation Baseline Surveys, Post Fuel Load Survey - field work complete on 12/12/15. No Acceptance or Review Criteria was associated with this procedure.
- 2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete for applicable sections on 12/16/15 with all criteria met.
- 2-PAT-5.1, Dynamic Automatic Steam Dump Control, field work complete for applicable sections on 1/16/16. There was no Acceptance or Review Criteria for this portion of testing.
- 2-PAT-3.10, Reactor Trip System, field work complete on 1/20/16 with all criteria met.
- -2-PAT-3.1, Control Rod Drive Mechanism and CERPI Initial Calibration, field work complete on 1/24/16 with all Acceptance Criteria met after evaluation of current amplitudes on twelve lift coils determined the results to be acceptable for the designed operation of the rod control system. CR 1128950 was written for high current amplitudes and closed following Westinghouse evaluation that determined the measurements to be acceptable. There was no Review Criteria for this PAT.

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0) (continued)

- 2-PAT-3.8, Rod Drop Time Measurement and Stationary Gripper Release Timing, Mode 5 Performance, field work complete for applicable sections on 1/24/16 with all Acceptance Criteria met. CR 1128964 was written due to the RDTC plots for each rod were inverted from the expected response. This did not impact performance of the test and was resolved prior to the Mode 3 performance. There was no Review Criteria for this test.
- 2-PAT-1.4, Pipe Vibration Monitoring, field work complete for applicable sections on 2/1/16 with all criteria met.

PAT testing on the plant primary side was suspended on 1/26/16 until plant conditions and surveillance completions allowed further testing. Condensate was placed on modified long cycle which allowed the completion of the applicable portions of 2-PAT-1.4.

On 3/15/16 preparations began for entering Mode 4 and PAT Test Coordinators began reviewing and completing pre-requisites for Mode 4 testing. Mode 4, RCS temperature >200°F and <350°F, entry was made on 3/19/16.

The 250°F Plateau included the following:

- 2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable section on 3/21/16 with all criteria met. There is no Review Criteria associated with this PAT.

The 300°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable sections on 3/25/16 with all criteria met.
- The plant entered Mode 3, RCS temperature ≥ 350°F, on 3/30/16 at 23:14 to allow further Power Ascension Testing.

The 360°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable sections on 3/31/16 with all criteria met.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable section on 3/31/16 with all criteria met.
- 2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete for applicable sections on 3/31/16 with all criteria acceptable for continued heat-up.

Plant heat-up to 400 degrees was initiated on 4/1/16 at 01:14.

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0) (continued)

The 400°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable sections on 4/1/16. CR 1156311 documented some test data was not taken with the RCP start but was collected satisfactorily from the plant computer. Review criteria was not met on the Reactor Coolant Pump Combination testing, The RVLIS system was updated with the new constants supplied by Westinghouse to correct the abnormality and documented in CR 1156425.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable sections on 4/1/16 with all criteria met.

After completion of the 400°F Plateau testing, plant heat-up to 450°F was initiated at 09:45 and completed at 12:16 on 4/1/16.

The 450°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable sections on 4/1/16 with all criteria met.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for applicable section on 4/1/16 with all criteria met.
- 2-PAT-1.8, Thermal Expansion of Piping Systems - field work complete for applicable sections on 4/1/16. Problem Report #1 was initiated within the test for seven snubbers not performing as expected. Results indicated no issue with the snubbers and approval was received to continue to next plateau testing.

Due to plant issues concerning check valve leakage, the decision was made on 4/2/16 to cool down the RCS and make entry into Mode 4 to allow repairs. Mode 4 entry was made on 4/2/16 at 06:29. Additionally, repairs on two RCS RTDs were made.

Mode 3 re-entry was made on 4/8/16 at 12:44.

Plant condition of RCS temperature at 500°F was met on 4/10/16.

The 500°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable sections on 4/10/16 with all criteria met.

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0) (continued)

Plant heatup to 557°F was completed at 15:30 on 4/13/16 with normal operating pressure reached at 01:30 on 4/14/16. On 4/17/16 at 03:38 the unit was placed in Mode 4 for repairs to the Auxiliary Feedwater Pumps and replacement of PD07-2 shim determined to require adjustment. The Unit was returned to Mode 3 on 5/1/16 at 17:36 and normal operating temperature and pressures on 5/2/16 at 23:00.

The 557°F Plateau included the following:

- 2-PAT-1.11, RVLIS Performance Test, field work complete for applicable section of 557°F data taking only on 4/13/16 with all criteria met for steady state data collection. Additionally, Section 6.1.3, Pump Combinations at 557°F, was field work complete on 5/8/16. Results (Section 6.1.4) indicated Acceptance Criteria would not be met. The system was updated with the new RVLIS constants supplied by Westinghouse to correct the abnormality and documented in CR 1171130.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, field work complete for Data Collection Section 6.7 on 4/13/16 with all criteria met. Section 6.8, Pump Contact Data Collection at 557°F was completed on 5/8/16 with all criteria met.
- 2-PAT-1.8, Thermal Expansion of Piping Systems, field work complete for applicable sections on 4/14/16 with one issue outside containment on PD07-2 which also required evaluation of PD07-1. Problem Report #2 was initiated to resolve the issue with PD07-2 and investigate any possible issues with PD07-1. Additionally Problem Report #3 was written to evaluate components not moving as expected. Both Problem Reports were closed and conditions were acceptable to continue testing.
- 2-PAT-1.4, Pipe Vibration Monitoring, field work complete for Section 6.5.1 Pressurizer Surge - Mode 3, on 4/14/16 with all criteria met for that section. Section 6.5.2, Main Feedwater Pump 2A Start and Steady State Operation on Recirc., was field work complete on 4/13/16 with velocity and displacement Acceptance Criteria not met. CR 1161783 was initiated for an engineering evaluation which concluded acceptable as is. Section 6.5.3, Main Feedwater Pump 2B Start and Steady State Operation on Recirc., was field work complete 5/5/16 with steady state velocity and displacement exceeding the Acceptance Criteria. CR 1168287 was written for an engineering evaluation and resulted in adjustment of a loose hanger and a retest. The retest was completed on 5/13/16 with satisfactory results. Section 6.5.4, Turbine Bypass Valve 2-FCV-1-105 Transient was completed on 5/12/16 with all criteria met. Section 6.5.5, Turbine Bypass Valve 2-FCV-1-111 Transient was completed on 5/12/16 and re-tested on 5/13/16. Engineering evaluation of the retest indicated satisfactory results.

CR 1170319 documents engineering evaluation to accept-as-is following the retest. Section 6.5.7 Condensate - Long Cycle was field work complete on 2/1/16 with all criteria met. There was no Review Criteria for 2-PAT-1.4.

- 2-PAT-3.2, Pressurizer Spray Capability and Continuous Spray Flow Setting - Section 6.1, Adjustment of the Pressurizer Manual Spray Bypass Valves, was completed on 4/16/16. All Acceptance Criteria was met. Review criteria for MCR alarms was not met with CRs 1161382 and 1160969 written. A Westinghouse evaluation determined the PAT met the operability and design requirements for the pressurizer spray system. Additionally, CR 1161789 was written for proportional heater band not within the specified requirement which was not an Acceptance Criteria.
- 2-PAT-3.3, RCS Flow Measurement, field work complete on 5/3/16 with all criteria met.
- 2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, field work complete on 5/6/16 with all Acceptance Criteria met. One Review Criteria was not met and CR 1168641 was initiated.
- 2-PAT-1.6, Startup Adjustments of Reactor Control System, field work complete for Mode 3 on 5/4/16. This performance was data taking only.
- 2-PAT-3.4, Rod Control and Rod Position Indication (CERPI), field work complete on 5/13/16. The Acceptance Criteria was not met in Sections 6.4 and 6.10. CRs 1168845, 1168881, and 1169602 were written to document failure to meet criteria. The criteria was re-evaluated and it was determined the acceptance criteria should be changed to require each Rod Position Indication to indicate rod motion consistent with the group demand indication for the full range of rod travel. A change to the Westinghouse Acceptance Criteria and SAR Change Package No. U2-019 were approved and an urgent change to the procedure incorporated the revised Acceptance Criteria. All Acceptance Criteria for the final package were met.
- 2-PAT-3.0, Attachment 1,Testing Deferred from 2-PTI-062-03, - field work complete on 5/7/16 with all Acceptance Criteria met. CR 1168487 was written to document alternate charging flow was not within anticipated range, however, it had no affect on the test acceptance.
- 2-PAT-3.8, Rod Drop Time Measurement and Stationary Gripper Release Timing, field work complete on 5/11/16. CR 1169659 was written for two rods failing a two sigma statistical evaluation. Three additional rod drops were performed and all Acceptance Criteria was met. There was no Review Criteria for this test.

- 2-PAT-3.11, Adjustment of Steam Flow Transmitters at Minimal Flow, field work complete on 5/7/16 with all Review Criteria met. There was no Acceptance Criteria associated with this performance.
- 2-PAT-3.7, Reactor Coolant Flow Coastdown, field work complete on 5/8/16 with all criteria met. CR 1169224 was written to document during removal of an instrument recorder a blown fuse caused alarms in the Main Control Room.
- 2-PAT-5.1, Dynamic Automatic Steam Dump Control, field work complete on 5/12/16 with all criteria met after a volume booster adjustment with CR 1170159 and retest on 2-FCV-1-108.

3.0 Test Results

Acceptance/Review Criteria were contained within the test sequenced by this test, except for Attachment 1, Testing Deferred from 2-PTI-062-03. Attachment 1 required Acceptance Criteria were met as delineated below.

Acceptance Criteria

3.1 Sum of RCP seal injection flow ≤ 40 gpm, (6-13 gpm for each RCP).

A. 2-FI-62-1A = 9.0 gpm (6.6-12.4 gpm)
B. 2-FI-62-14A = 9.2 gpm (6.6-12.4 gpm)
C. 2-FI-62-27A = 9.3 gpm (6.6-12.4 gpm)
D. 2-FI-62-40A = 9.2 gpm (6.6-12.4 gpm)

Total Seal Injection Flow Rate in gpm. = A + B + C + D =
9.0+ 9.2 + 9.3 + 9.2 = 36.7 gpm

3.2 The differential pressure across the following component at the given flowrate:

Description	UNID	Flow Rate	ΔP (Clean)	Actual ΔP
Seal Injection Filter B	2-FLTR-62-96	16-40 gpm	≤7 psid	6.0 psid

3.3 Indication light 2-XI-62-93 in MCR illuminated when 2-HIC-62-93B was in manual.

5.1 Post Core Loading Precritical Test Sequence (2-PAT-3.0) (continued)

3.4 2-FM-62-93E prevented 2-FCV-62-93 from going fully closed to ensure Seal Water flow rate of 33.5 ±1.5 gpm (32-35 gpm). Flow was 32.2 gpm.

Review Criteria

None

4.0 Problems

[1] CR 1168487 was written on 2-PAT-3.0, Attachment 1, Step 27. Although not Acceptance Criteria, alternate charging header flow was anticipated to be approximately 89-103 gpm. Actual flow was 82.7 gpm. This information was forwarded to engineering for evaluation, however, it had no affect on the acceptance of this test. Engineering evaluation calculated the minimum requirement at the regenerative heat exchanger temperature to be 74 gpm. The 82.7 gpm exceeds this amount. At the current conditions the test was acceptable and met the design specified criteria.

Additional problems encountered are addressed in the following discussions of each test sequenced by 2-PAT-3.0.

5.2 Control Rod Drive Mechanism Timing and CERPI Initial Calibration (2-PAT-3.1)

Performance of this test was directed by 2-PAT-3.0, Post Core Loading Pre-critical Test Sequence, during the period from 1/21/16 to 1/24/16. The test was performed in Mode 5 at a RCS temperature of approximately 175°F.

1.0 Test Objectives

The objectives of this test were to:

1.1 Verify the functionality of each CRDM for shutdown and control rods in Mode 5 by:

1.1.1 Verify each rod control system slave cycler provides its associated power cabinet with the appropriate command signal to obtain proper sequence timing of current supplied to the CRDM coils.

1.1.2 Verify CRDM coil current amplitudes are within acceptable ranges.

1.1.3 Verify the functionality of each shutdown and control rod drive mechanism.

1.1.4 Verify manual mode stepping rate for shutdown and control rods are within acceptable ranges.

1.2 Verify the control bank overlap function in manual with minimal overlap.

1.3 Perform the initial calibration of the RPI in accordance with vendor procedure WNA-TP-02576-WBT, CERPI Calibration Procedure.

1.4 Partially satisfy the requirements of UFSAR Table 14.2-2, Sheet 7, Control Rod Drive Mechanism Timing Test Summary, and fully satisfy it for Mode 5.

2.0 Test Methods

The CRDM functionality was verified by stepping out/in each rod bank by approximately 10 steps in individual bank select mode. CRDM current timing and amplitude measurements were taken during rod motion. Twelve of the CRDM amplitudes were outside the upper Acceptance Criteria of the lift coil reduced current, however, the amplitudes were evaluated as acceptable by Westinghouse.

5.2 Control Rod Drive Mechanism Timing and CERPI Initial Calibration (2-PAT-3.1) (continued)

The bank overlap circuitry was verified at minimal settings. Minimal settings were set by adjusting the bank overlap thumbwheel switches such that control bank tip-to-tip distance was 15 steps and the all-rods-out position was 25 steps withdrawn for each control bank. The bank overlap circuitry functioned as designed and no issues were encountered. Note that the bank overlap circuitry was also exercised during the performance of the initial RPI calibration with the all-rods-out position set to 230 steps withdrawn. The bank overlap circuitry functioned as designed with no issues.

The initial RPI calibration was performed. First the bank zero adjustments were performed with all rods fully inserted. Next all shutdown and control rods were withdrawn to the full out position of 230 steps withdrawn. The shutdown banks were withdrawn first in individual bank select and the control rods were withdrawn in bank overlap. The bank position span calibration and temperature null adjustments were performed with the rods fully withdrawn. Next all control and shutdown rods were inserted to specific demanded positions and data for each rod was obtained. Lastly, linearization adjustments were calculated based on the recorded data. The initial RPI calibration was completed when the new linearization adjustments were uploaded to train A and B of the RPI system. Note that prior to the linearization adjustment and during the insertion of Control Bank B, the K14 and P6 rods had a 13 step rod-to-rod deviation while inserted between CBB demanded positions of 170 steps withdrawn to 126 steps withdrawn. The RPI system was not yet calibrated, therefore, the initial calibration corrected the issue. Also note that 2-PAT-3.8 was performed following the initial RPI calibration and all rods were pulled to approximately 50 steps withdrawn. All rods were within ±2 steps of the demanded position.

3.0 <u>Test Results</u>

All Acceptance/Review Criteria were met or resolved as delineated below.

<u>Acceptance Criteria</u>

3.1 Control Rod Drive Mechanism Timing

3.1.1 Current Order Timing

The times at which the lift, movable, and stationary current orders change, after the start of rod motion, are within 10 msec. of the expected times during rod withdrawal and insertion operations.

5.2 Control Rod Drive Mechanism Timing and CERPI Initial Calibration (2-PAT-3.1) (continued)

Each CRDM current order timing was reviewed and all current order timings were within 10 msec. of the expected times.

3.1.2 Coil Current Amplitudes

Stationary, movable, and lift currents are regulated by circuitry internal to each power cabinet. The reduced and full current nominal values are not critical, cannot be adjusted, but could be an indication of a regulation failure. Measured values outside the nominal ranges should be evaluated and documented by the system engineer.

Lift Coil - Full	Nominal 40 amperes (35 to 47.2 amperes) (equivalent to 438 to 590 mVdc measured across a 0.0125 ohm resistor)
Lift Coil - Reduced	Nominal 16 amperes (13 to 19.7 amperes) (equivalent to 163 to 246 mVdc measured across a 0.0125 ohm resistor)
Movable Gripper Coil - Full	Nominal 8 amperes (7 to 9.2 amperes) (equivalent to 438 to 575 mVdc measured across a 0.0625 ohm resistor)
Stationary Gripper Coil - Full	Nominal 8 amperes (7 to 9.2 amperes) (equivalent to 438 to 575 mVdc measured across a 0.0625 ohm resistor)
Stationary Gripper Coil - Reduced	Nominal 4.4 amperes (3.8 to 4.8 amperes) (equivalent to 238 to 300 mVdc measured across a 0.0625 ohm resistor)

Each current amplitude recorded in the test package were reviewed. All current amplitudes were within the Acceptance Criteria with the exception of twelve CRDMs for the lift coil reduced current. The twelve lift coil reduced current amplitudes were evaluated and determined to be acceptable for the designed operation of the rod control system. CR 1128950 was closed.

3.1.3 Rod Withdrawal Speed

Shutdown bank withdrawal speed nominal value is 64 (62 to 66) steps per minute.

Control bank withdrawal speed nominal value is 48 (46 to 50) steps per minute.

The Shutdown and Control banks withdrawal speeds met their Acceptance Criteria and were recorded as 63.9 steps/min and 48.0 steps/min respectively.

3.1.4 Rod Drive Mechanism Operability

Shutdown rod drive mechanisms operate with no indications of problems during the withdrawal and insertion stepping.

Control rod drive mechanisms operate with no indications of problems during the withdrawal and insertion stepping.

Each CRDM current trace was reviewed. All traces operated normally and no abnormalities, such as movable/stationary gripper dragging or rod misstepping, were identified.

3.2 Control Bank Overlap Demonstration

3.2.1 The control rod bank overlap circuitry functions properly during the sequential withdrawal and insertion of Control Banks in MANUAL mode.

The control bank overlap circuitry functioned as designed.

3.2.2 The MCR rod speed demand display functions properly and indicates the rod stepping rate was within the range of 46 to 50 steps/minute for Control Banks in Manual mode.

The MCR rod speed demand display functioned as designed at 48.0 steps/min.

3.2.3 The MCR group step counters function properly to indicate group position and direction of rod motion during rod withdrawal and insertion operations.

The MCR group step counters functioned as designed.

3.2.4 The MCR RPI functions properly to indicate individual rod direction of motion during rod withdrawal and insertion operations.

The MCR RPIs indicated the proper direction of motion during rod withdrawal and insertion operations.

3.2.5 The MCR rod direction indicator lights function properly to indicate the rod movement status and direction of rod motion during rod withdrawal and insertion operations.

The MCR rod direction indicator lights functioned as designed.

<u>Review Criteria</u>

None

4.0 Problems

[1] During the performance of the CRDM timing and amplitude measurements, the 1AC power cabinet- stationary group A coil amplitudes were lower than the expected value. WO 117522924 was performed to inspect and reform backplane connector and card edge pins for the 1AC power cabinet - stationary group A firing, regulation, and phase control cards. The issue was corrected and testing continued.

5.2 Control Rod Drive Mechanism Timing and CERPI Initial Calibration (2-PAT-3.1) (continued)

[2] Step 6.3.3[7.1], WNA-TP-02576-WBT, Revision 2, Step 2.4.2.5, the all-rods-out position was 230 steps withdrawn, however, the compensated position in the software was hardcoded to 231 steps withdrawn. The performance of Step 2.4.2.5 was not impacted because it listed 230 steps ±1 step. CR 1128373 was written and concluded no changes to 2-PAT-3.1 were required.

[3] CR 1128918: Step 6.3.4[62], WNA-TP-02576-WBT, Revision 2, Appendix A.1 and A.2 forms were used for the linearization adjustments. The "X Table C1" column values were not Watts Bar U2 specific values. The Watts Bar U2 plant specific "X Table C1" values were used for the linearization adjustments. CR 1128918 was written. Resolution was for Westinghouse to revise WNA-TP-02576-WBT.

[4] CR 1128950 Two rod amplitude measurements (F14 and D08) failed the procedure Acceptance Criteria and do not meet the Westinghouse expanded acceptance criteria in WBT-D-5420.

Westinghouse has provided a letter (WBT-D-5604(3.8)) documenting their evaluation and the acceptability for the 2 rod locations that exceeded the Acceptance Criteria of 20 amps.

Note that additional rod measurements were outside of the Reduced Lift Current procedure Acceptance Criteria of 19.7 amps, however, WBT-D-5420 has been issued by Westinghouse that states reduce lift currents up to 20.0 amps are acceptable. 2-PAT-3.1, Rev. 2, allows for evaluation of the currents outside of the Acceptance Criteria for successful completion of 2-PAT-3.1.

Current orders outside of the Acceptance Criteria were evaluated and deemed acceptable per Westinghouse. Also additional measurements were obtained in Mode 3.

[5] During the performance of 2-TRI-85-1, Reactivity Control Systems Movable Control Assemblies (Modes 3, 4 and 5), rods common to the 2BD power cabinet would not withdraw. Troubleshooting determined an issue with the 2BD movable gripper current amplitudes. The firing and regulation cards for the 2BD movable grippers were replaced with spares and the issue was corrected. No problems with this power cabinet occurred during 2-PAT-3.1 testing.

5.2 Control Rod Drive Mechanism Timing and CERPI Initial Calibration (2-PAT-3.1) (continued)

Step 6.1[12] of Appendix E, rods common to the 1AC power cabinet - stationary group A did not have the expected reduced stationary gripper currents when the CRDM-DAQ was first connected. Under CR 1126661 and WO 117522924, the stationary group A firing, regulation, and phase cards were removed and both backplane connector and card edge pins were reformed. The issue was corrected and testing was completed. No other problems with this power cabinet occurred during 2-PAT-3.1 testing.

[6] During the performance of 2-TRI-85-1 with the 2-RBSS in the SBC and SBD positions, the CERPI monitor indicated 72 steps/min. The actual speed of the SBC and SBD groups is approximately 64 steps/min and is set at the SCD power cabinet. The indication did not invalidate the performance of this test. CR 1126783 was written and closed to WO 115966328.

[7] Step 6.3.4[3], during SBD insertion, the SBC demand position on both the ICS and RPI monitors followed the SBD demand position. The problem was due to communication issues between the rod control system and the ICS. Testing continued because this issue did not invalidate the performance of this test. CR 1128318 was written, and Post Issuance Change (PIC) 66181 was issued and implemented by WO 117546244.

[8] Step 4.3[10]A, the 2B MG Set failed to sync in parallel with the 2A MG Set. The issue did not invalidate the performance of 2-PAT-3.1 because only one MG set was required for the performance of this test. CR 1126798 was written, and closed to WO 117531764.

5.3 Pressurizer Spray Capability and Continuous Spray Flow Setting (2-PAT-3.2)

Performance of this test was directed by 2-PAT-3.0, Post Core Loading Pre-critical Test Sequence, during the period from 4/15/16 to 5/4/16.

1.0 <u>Test Objectives</u>

The objectives of this test were to:

1.1 Verify the pressure response to the opening of both normal Pressurizer Spray Valves was within the allowable range specified by NSSS performance curves.

1.2 Verify the Pressurizer Bypass Spray Valves were throttled to an optimum position such that during steady state operation:

1.2.1 Spray line temperature was high enough to prevent the PZR SPRAY TEMP LO alarm from actuating.

1.2.2 The equilibrium temperature for each spray line was greater than or equal to 540F.

1.2.3 Pressurizer control bank heaters can maintain RCS pressure above 2220 psig without backup heater operation.

1.2.4 Surge line temperature was high enough to prevent the PZR SURGE LINE TEMP LO alarm from actuating.

1.3 Verify the PZR SPRAY TEMP LO alarm would actuate on decreasing spray line temperature of approximately 530F.

1.4 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 13, Pressurizer Spray Capability And Continuous Spray Flow Setting Test Summary.

5.3 Pressurizer Spray Capability and Continuous Spray Flow Setting (2-PAT-3.2) (continued)

2.0 Test Methods

This test established the optimal throttle positions for the Pressurizer Spray Manual Bypass Valves, and also ensured the effectiveness of the normal pressurizer spray by initiating full spray to reduce RCS pressure by approximately 250 psi and compared the time to reduce pressure with Westinghouse performance curves. During the performance of Section 6.2, the validated RCS pressure DCS computer point being used to monitor the depressurization of the RCS stopped updating at acceptable rate. Because of this the RCS narrow range indicators on the control board were used to determine when RCS pressure reached the trigger value of 2000 psig. Subsequent review also determined that this point deviated further from the actual RCS pressure after it was no longer being monitored. This did not affect the ability to meet the Acceptance Criteria of the test as ICS computer points were collected for use to analyze compliance with Acceptance Criteria for the spray capability test. CR 1168255 documents this issue.

The spray line temperature low alarm was unable to be verified as intended during the performance of this test due to slight leakage past either the spray line FCV's or the spray bypass manual valves. This occurred on both loops 1 and 2. This condition prevented meeting the Review Criteria associated with the spray line temperatures. The operation of the spray line temperature switches for each loop were subsequently verified to be operating correctly by utilizing trend data from the plant computer. CRs 1160969 and 1161382 document this issue.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 Pressurizer pressure response to opening both Normal Pressurizer Spray Valves is within the allowable range specified by NSSS performance curves.

The pressurizer spray response data was within the allotted response time as depicted on Figure 5.3-1.

5.3 Pressurizer Spray Capability and Continuous Spray Flow Setting (2-PAT-3.2) (continued)

Review Criteria

3.2 Pressurizer Manual Spray Bypass Valves 2-BYV-68-552 and 2-BYV-68-555 are throttled to an optimum position during steady-state operation.

All procedural criteria was met:

1. Spray line equilibrium temperature is high enough to prevent Annunciator 2-XA-55-5A-89E, PZR SPRAY TEMP LO from actuating.

   2-XA-55-5A-89E, PZR SPRAY TEMP LO, did not actuate.

2. Equilibrium temperature for each spray line is greater than or equal to 550°F:
   Loop 1 Spray Line Temperature (ICS PID T0484A)
   Loop 2 Spray Line Temperature (ICS PID T0483A)

   Spray line equilibrium temperatures were ≥ 550°F.

3. Pressurizer control bank heaters can maintain RCS pressure above 2220 psig without any Backup Heater operation.

   Backup Heater operation was not required to maintain pressure.

4. Surge line equilibrium temperature is high enough to prevent Annunciator Alarm 2-XA-55-5A/89D, PZR SURGE LINE TEMP LO, from actuating.

   2-XA-55-5A/89D, PZR SURGE LINE TEMP LO, did not actuate.

3.3 Annunciator Alarm 2-XA-55-5A/89E, PZR SPRAY TEMP LO actuates on decreasing spray line temperature of approximately 530°F (525°F to 535°F).

Loop 1 spray line temperature would not decrease sufficiently to allow the low spray line temperature alarm to actuate. See CR 1160969.

Loop 2 spray line temperature would not decrease sufficiently to allow the low spray line temperature alarm to actuate. See CR 1161382.

5.3 **Pressurizer Spray Capability and Continuous Spray Flow Setting (2-PAT-3.2)** (continued)

4.0 Problems

[1] CR 1160969 was written because the loop 1 spray line temperature would not decrease sufficiently to allow the low spray line temperature alarm to actuate. Westinghouse evaluation was obtained which concluded that 2-PAT-3.2 met operability and design requirements for the pressurizer spray system.

[2] CR 1161382 was written because the loop 2 spray line temperature would not decrease sufficiently to allow the low spray line temperature alarm to actuate. Westinghouse evaluation was obtained which concluded that 2-PAT-3.2 met operability and design requirements for the pressurizer spray system.

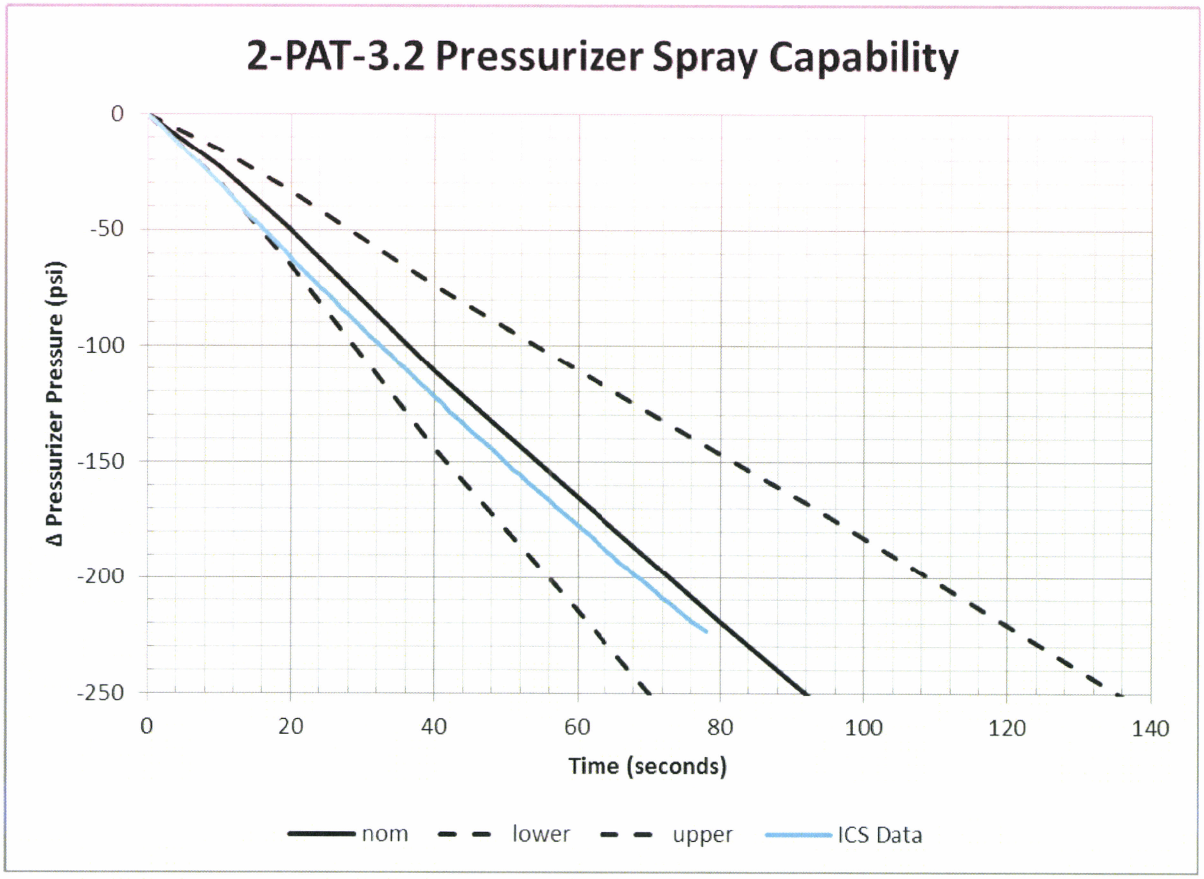
[3] CR 1161789 was written due to the master pressure controller output being less than the desired range while setting the loop 2 spray line bypass valve. The controller output was 37 percent while the desired range was a minimum of 46 percent. Performance of Section 6.1.3 which performed the final setting of the spray bypass valves, allowed the output of the master controller to be placed in the desired range of the procedure.

[4] CR 1168255 was written to document the issues experienced with the ICS computer point for Validated Pressurizer Pressure (DCS0426) during the performance of Section 6.2.

5.3 Pressurizer Spray Capability and Continuous Spray Flow Setting (2-PAT-3.2) (continued)

FIGURE 5.3-1

Pressurizer Spray Response



5.4 Rod Control and Rod Position Indication (CERPI) (2-PAT-3.4)

This test was performed in Mode 3 at NOTP as directed by 2-PAT-3.0, Post Core Loading Precritical Test Sequence. It performed the initial hot calibration of the Computer Enhanced Rod Position Indication (CERPI) system and functional testing of the Rod Control System. The performance of Section 6.0 of this test was commenced on 05/05/16 and was completed on 05/13/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Perform the Initial Hot Calibration of the Computer Enhanced Rod Position Indications (CERPI) system.

1.2 Verify the Computer Enhanced Rod Position Indication system (CERPI) performs required indication function satisfactorily for each shutdown and control rod over their entire range of travel and to verify the rod position indication system alarm functions operate properly. (UFSAR Table 14.2-2, Sheet 8, Rod Position Indication System Test Summary).

1.3 Demonstrate that the rod control system satisfactorily performs the required control and indication functions, as required by UFSAR Table 14.2-2, Sheet 10, Rod Control System Test Summary.

2.0 Test Methods

The rod position indication system completed the initial hot calibration using vendor instructions and 2-SI-85-3, Calibration of Computer Enhanced Rod Indication Channels and Full Range Verification. The CERPI system operated over the full length of travel and can operate without actuating rod-to-rod and rod-to-bank deviation alarms by making adjustments to the CERPI tunable parameters. This was consistent with vendor and Unit 1 operating experience. Therefore, the vendor CERPI Acceptance Criteria was revised and Urgent Change (UC) 2 was processed for this test procedure to verify that each rod indicates rod motion consistent with the group demand over the full length of travel.

The Rod Control System and CERPI functional testing included controls and indications. The functional testing included rod-to-bank and rod-to-rod deviation alarms, the C-11 annunciator, Integrated Computer System (ICS) generated alarms, rod bottom bistables, rod bottom bypass bistables, rod control urgent and non-urgent alarms, main control room displays, rod insertion limits, and control rod bank overlap circuitry.

This test also documented 5 complete rod excursions (i.e., full withdrawal and insertions) of all shutdown and control rods per CR 234483. All 5 excursions were successfully completed.

5.4 Rod Control and Rod Position Indication (CERPI) (2-PAT-3.4) (continued)

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 CERPI Calibration

3.1.1 WNA-TP-02576-WBT, Watts Bar 2 ARPI System Upgrade CERPI Calibration Procedure, Section 2.6, INITIAL HOT CALIBRATION, was successfully completed and the linearity is within ±12 steps at the steps checked in the procedure.

The initial hot calibration was successfully completed and the linearity was demonstrated to be ±12 steps at the steps checked in the procedure.

3.1.2 2-SI-85-3, Calibration of Computer Enhanced Rod Indication Channels and Full Range Verification, was successfully completed.

2-SI-85-3 was successfully completed.

3.2 Rod Control and Indication

3.2.1 2-XA-55-4A-64F, C11 BANK D AUTO WITHDRAWAL BLOCKED, alarm window in control room was LIT when Control Bank D was withdrawn above 220 steps.

2-XA-55-4A-64F, C-11 alarm window annunciated at 220 steps.

3.2.2 CERPI monitor alarm for:

(1) Rod to Rod Deviation between two rods in a bank ≥ 12 steps.

The rod to rod deviation CERPI alarm was at 12 steps.

(2) Rod to Bank Deviation corresponding to ≥ 12 steps.

The Rod to Bank Deviation alarm was at 12 steps.

3.2.3 2-XA-55-4B-83D, PLANT COMPUTER GENERATED ALARM (SEE ICS), alarm window in control room was LIT when ICS Computer detects the following conditions:

(1) Deviation between rod position indicator for a rod and corresponding bank demand position ≥12 steps.

The deviation between ICS rod position and bank demand was 12 steps.

(2) Deviation between rod position indicator for a rod and average rod position ≥12 steps.

The deviation between ICS rod position and average rod position was 13 steps.

3.2.4 Rod bottom bistable indicators actuate at correct setpoint setting (below 20 steps withdrawn) as indicated by RPI indicators and rod bottom indicators on CERPI on 2-M-4.

Each rod bottom bistable for all rods actuated below 20 steps (20 to 19 steps) withdrawn.

3.2.5 CERPI bypass indication for Control Banks B, C, and D actuate at correct setpoint setting (below 35 steps withdrawn) as indicated by CERPI Bank Demand digital display.

CERPI bypass indication for Control Banks B, C, and D actuated at 31 steps withdrawn.

3.2.6 An Urgent Failure induced in a Power Cabinet and Logic Cabinet caused local urgent failure alarm indicator lamp at the respective cabinet and 2-XA-55-4B-86A, CONTROL ROD URGENT FAILURE, annunciator window to light.

The Power Cabinet and Logic Cabinet local urgent failure alarm indicator lamp and 2-XA-55-4B-86A annunciator window functioned as designed and met all applicable Acceptance Criteria.

3.2.7 A Non-Urgent Failure induced in each of the Power Cabinets and Logic Cabinet causes local non-urgent failure alarm indicator lamp at respective cabinets and 2-XA-55-4B-86B, CONTROL ROD NON-URGENT FAILURE, annunciator window to light.

The Power Cabinet and Logic Cabinet local non-urgent failure alarm indicator lamp and 2-XA-55-4B-86B annunciator windows functioned as designed and met all applicable Acceptance Criteria.

3.2.8 A failure induced in the CERPI racks caused 2-XA-55-4B-86C, CERPI TROUBLE, annunciator window to be LIT or REFLASH.

The 2-XA-55-4B-86C annunciator window functioned as designed and met all applicable Acceptance Criteria.

3.2.9 For shutdown and control rod banks having two groups, the group step counter for group 1 shall be 0 or 1 step above group 2 step counter over their full length of travel (i.e., 231 steps).

The group step counters for group 1 were 0 or 1 step above group 2 step counters over their full length of travel for shutdown and control rod banks having two groups. All applicable Acceptance Criteria were met.

3.2.10 2-XA-55-4B-87D, RODS AT BOTTOM, annunciator window was lit when one or more rods in CBA were inserted in the normal sequence. Also, RODS AT BOTTOM, annunciator window is not lit when control rods were inserted or withdrawn in their normal sequence.

2-XA-55-4B-87D annunciator window functioned as designed and met all applicable Acceptance Criteria.

3.2.11 Each RPI indicated rod motion consistent with the group demand indication for the full range of rod travel.

The RPI indicators for each rod indicated rod motion consistent with the group demand indication for the full range of rod travel.

5.4 Rod Control and Rod Position Indication (CERPI) (2-PAT-3.4) (continued)

3.2.12 The MCR rod speed demand display functions properly and indicated the rod stepping rate (ROD SPEED) was within the range of 62 to 66 steps/minute for Shutdown Banks A and B in bank select mode.

The MCR rod speed demand display indicated 64 steps/min for both Shutdown Banks A and B.

3.2.13 The MCR rod speed demand display functions properly and indicates the rod stepping rate (ROD SPEED) was within the range of 46 to 50 steps/minute for Control Banks in bank select and in MANUAL mode.

The MCR rod speed demand display indicated 48 steps/min for all Control Banks in bank select and in Manual mode.

3.2.14 The MCR rod direction indicator lights functioned properly to indicate the rod movement status and direction of rod motion during rod withdrawal and insertion operations.

The MCR rod direction indicator lights functioned as designed and met all applicable Acceptance Criteria.

3.2.15 The MCR group step counters functioned properly to indicate group position and direction of rod motion during rod withdrawal and insertion operations.

The MCR group step counters as designed during both withdrawal and insertion.

3.2.16 The MCR Computer Enhanced Rod Position Indicators (CERPI) function properly to indicate individual rod position and direction of motion during rod withdrawal and insertion operations.

The CERPI indicators functioned as designed to indicate individual rod position and direction of motion during withdrawal and insertion. All applicable Acceptance Criteria were met.

3.2.17 The rod insertion limits LO-LO upper limit was set to 211 steps (Control Bank A)

The rod insertion limits LO-LO upper limit was found to be set at 211 steps (Control Bank A).

3.2.18 The rod insertion limits LO alarm actuated below 10 steps above the insertion limit for any control bank.

The rod insertion limits LO alarm actuated at 10 steps above the insertion limit for any control bank.

3.2.19 The rod insertion limits LO-LO alarm actuated below 0 steps above the insertion limit for any control bank.

The rod insertion limits LO-LO alarm actuated at 0 steps above the insertion limit for any control bank.

3.2.20 The control rod bank overlap circuitry functioned properly during the sequential withdrawal and insertion of Control Banks in MANUAL mode.

The control bank overlap circuitry functioned as designed during the sequential withdrawal and insertion of Control Banks in MANUAL mode.

3.2.21 Each RPI indicated rod motion consistent with the group demand indication for the full range of rod travel.

The RPI indictors for each rod indicated rod motion consistent with the group demand indication for the full range of rod travel.

3.2.22 All rods were fully withdrawn and inserted five times.

The test exercised all rods to fully withdrawn and fully inserted five times.

Review Criteria

None

4.0 Problems

[1] CR 1168845 - Steps 6.4.3[20] and 6.4.4.[20], Acceptance Criteria for RPI indication agreeing within ±12 steps was not satisfied for Shutdown Banks C and D for the full length of travel. This Acceptance Criteria was later changed by Urgent Change 2. See UC-2 description in this test report for further information.

[2] CR 1168881 - Section 6.4.5, Rod position indications for rods in Control Banks A and B were at 12 steps from the demand position at some positions over the full length of travel. Although not a failure in the Acceptance Criteria of Step 6.4.5[20] of within 12 steps of the demand position, the rod-to-rod deviation alarms were actuated at certain positions over the length of travel. This CR anticipated not meeting Steps 6.10[2.15] and 6.10[2.16]. The acceptance of rod-to-rod deviations was later changed by Urgent Change 2. See UC-2 description in this test report for further information.

[3] CR 1169602 - Steps 6.10[2.15] and 6.10[2.16] Acceptance Criteria of no rod-to-rod and rod-to-bank deviation alarms was not met for all 57 rods over the full length of travel. These steps were later changed by Urgent Change 2. See UC-2 description in this test report for further information.

[4] CR 1169217 - Step 6.4.6[1]C verified the C-11 Bank D Auto Withdrawal Block annunciator cleared during Control Bank D insertion between 219 and 214 steps on the step counters. The recorded step counter position for Control Bank D was 212 steps when the C-11 annunciator cleared. Step 6.4.6[1]C is not Acceptance Criteria and the value at which the annunciator cleared is reasonable. Based on the CR evaluation, the vendor manual description of operation demonstrates (as well as the test engineer evaluation) there was not a problem encountered during 2-PAT-3.4 performance with Control Rod Bank D permissive C-11. No Further Action Required.

[5] CR 1169282 - Step 6.7.2[7] could not be performed as written because the performance of Step 6.7.2[6] cleared the urgent alarm upon seating the A104 card. The card interlock is the only urgent alarm in the rod control system that clears upon restoring the system configuration. Step 6.7.2[7] was to verify rod motion would not occur with a standing urgent alarm. These steps were not Acceptance Criteria and had no impact on successful completion of this test. Additionally, CR 1171247 was created to address and provide justification associated with not performing the steps cited in CR 1169282.

[6] CR 1171254 - This CR was written to document certain steps and portions of sections which were repeated during testing performance based on engineering judgment. These additional performances were used as a means to perform additional calibration of CERPI as specified in the Westinghouse CERPI calibration procedure or as a means to restore from current testing

conditions and then later return. Section 6.1, Steps 6.1[1] through 6.1[11.19] were repeated and Section 6.6, Steps 6.6[1] through 6.6[9.5] were repeated one or more times. Section 6.6, Steps 6.6[48] through 6.6[54] were repeated.

This CR was created for documentation purposes only and has no impact on the actual test results or verification of Acceptance Criteria.

[7] CR 1168538 - During performance of Section 6.1, Shutdown Bank A was inserted to 118 steps demand position for CERPI hot calibration. M14 indication drifted excessively for a couple of hours until a stable rod position indication was reached.

The PAT team and Westinghouse reviewed the M14 coil resistance values identified that the M14 drift was associated with a large transient in coil resistance (i.e., change in coil stack temperature) and a larger value for the T_GAIN parameter for M14.

Drift is a phenomena that occurs for ARPI/CERPI indication systems due to the analog coil stacks and their associated temperature dynamics and does not represent an actual change in rod position. This does not represent a deficiency in the design. Adjustments to the CERPI tunable parameters were made to minimize drift. Therefore, this CR did not impact the successful completion of this test procedure.

[8] CR 1168899 - Step 6.4.5[8], the 2-XA-55-4B-87D, RODS AT BOTTOM, annunciator cleared as expected. However, the bell in the MCR did not alarm. CR was closed to previously identified work.

[9] UC-1 was written to ensure that group step counters for Shutdown Bank A, group 1, step counters displayed 56 steps prior to withdraw of Rod D-02 in Step 6.6 [40]. This allowed the group step counter to be updated to match the current rod position for rod that was currently capability of motion.

[10] UC-2 was processed to revise Acceptance Criteria 5.2K and 5.2U and related steps based on updated criteria provided by Westinghouse in WBT-D-5666, CERPI Acceptance Criteria, Revision 1, and a revision to UFSAR Table 14.2-2, Sheet 8. The update criteria stated "Each RPI indicates rod motion consistent with the group demand indication for the full range of rod travel."

5.5 Reactor Coolant Flow Coastdown (2-PAT-3.7)

This test was performed as directed by 2-PAT-3.0, Post Core Loading Precritical Test Sequence, in Mode 3 at normal operating temperature and pressure. The test was started and field work completed on 5/8/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Measure the rate at which reactor coolant flow changes subsequent to a simultaneous trip of all four reactor coolant pumps. The measured Flow Coastdown Time Constant is determined from the flow versus time data and compared to the Design Flow Coastdown Time Constant.

1.2 Measure the delay time associated with the low flow reactor trip and compare it to that value assumed in the accident analysis.

1.3 Record the RCP Motor voltage decay during the transient for information only.

1.4 This test satisfied the requirements of UFSAR Table 14.2-2, Sheet 15, Reactor Coolant Flow Coastdown Test Summary.

2.0 Test Methods

All four reactor coolant pumps were simultaneously tripped, causing the reactor trip breakers to open on Low RCS Flow. Measurements were made by recording reactor coolant loop elbow tap differential pressures (d/p), RCS low flow bistable state, reactor trip breaker position, reactor coolant pump breaker position and reactor coolant pump motor voltage decay data. Also recorded for information was the time of the undervoltage relay and associated time delay timer in the RCPs undervoltage circuit.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 The Acceptance Criterion for core flow coastdown following the simultaneous trip of the four reactor coolant pumps from full flow conditions, was that the measured flow coastdown time constant (TAU_M) was greater than (>) design flow coastdown time constant (TAU_D) of 11.72 seconds.

Results indicated TAU_M = 12.762 seconds

3.2 Acceptance criterion for the Total Low Flow Trip Delay Time is less than (<) 1.2 seconds.

Results indicated Low Flow Trip Delay Time, T_{LF} = 0.994 seconds

3.3 Acceptance criterion for simultaneous trip of four reactor coolant pumps was that all four pumps trip within (≤) 100 msec. of each other.

All four pumps tripped within 20 msec.

Review Criteria

3.4 Review Criterion for coastdown flow data quality is that data from at least 2 out of 3 flow transmitters in each RCS loop falls within Chauvenet's Criterion.

All data points from all RCS flow transmitters fell within Chauvent's Criterion.

4.0 Problems

[1] CR 1169224 - During performance of 2-PAT-3.7 post performance activities, Step 7.0[5], removing the test recorders from the Aux Instrument Room, 8 out of 12 RCS low flow trip status lights lit in the control room. Post event investigation revealed bistable fuses associated with the affected flow loops (2-LPF-68-6A, 2-LPF-68-6B, 2-LPF-68-29A, 2-LPF-68-29B, 2-LPF-68-29D, 2-LPF-68-48B, 2-LPF-68-71A, 2-LPF-68-71B) blew during recorder disconnection because of human performance issues.

Since this was a post performance activity and all data was recorded during Section 6.0, there was no impact to the test results of 2-PAT-3.7.

5.6 Rod Drop Time Measurement and Stationary Gripper Release Timing (2-PAT-3.8)

Portions of this test were performed in Mode 5 and again in Mode 3 as directed by 2-PAT-3.0, Post Core Loading Precritical Test Sequence.

In Mode 3 this test was performed in conjunction with the normal Surveillance Instruction 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer, to calculate the standard deviation of the rod drops and to direct required additional rod drops for CR corrective actions and potential two-sigma deviations.

Prerequisites were started on 1/22/16 and the test was field work completed on 1/24/16 for the Mode 5 performance. The Mode 3 performance was started on 5/11/16 and completed on 5/11/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 In Mode 5, Section 6.2 of this Power Ascension Test (PAT) partially withdrew all shutdown and control rods and demonstrated that all CRDMs unlatch and all rods fully insert into the core when the reactor trip breakers were opened.

1.2 In Mode 5, measured the Stationary Gripper Release times for each control and shutdown rod.

1.3 In Mode 3, at Hot Standby conditions with full Reactor Coolant System (RCS) flow, measured the rod drop time and stationary gripper release time for each control and shutdown rod. WBN Unit 2 Technical Specifications require rod drop time measurements, therefore, the normal Surveillance Instruction 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer was utilized.

1.4 Meet the Mode 3 testing as required by UFSAR Table 14.2-2, Sheet 9, Rod Drop Time Measurement And Stationary Gripper Release Timing Test Summary

- Measure the stationary gripper release time for each control and shutdown rod. This measurement was performed in Mode 5 and then repeated in Mode 3.

- Evaluate the data from rod drop time testing in the area of the dash pot entry looking for proper performance of the decelerating devices (i.e. dashpots). This evaluation was performed in Mode 3 with the data collected during the performance of 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer.

- Evaluate all 57 rod drop times in Mode 3 with the data collected during the performance of 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer.

1.5 Ensure that four rod drops were performed in Mode 3 as required by corrective actions from Condition Report 234483 action 003 related to INPO SER 1-10.

2.0 Test Methods

This PAT was written to supplement the normal operating surveillance 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer and evaluated the rod drop time data.

All rod drop times were used to calculate the standard deviation of the rod drop times. Two-sigma limits (i.e. plus or minus two times the standard deviation) were used to evaluate drop times of the 57 rods. Those drop times that were outside of the two-sigma limits were re-measured 3 (or more) times and evaluated for consistency (i.e. within 50 milliseconds). Retesting the rods that fell outside of the two-sigma limits an additional 3 (or more) times provided reasonable assurance of their proper performance during subsequent plant operations.

This PAT measured the Stationary Gripper Release Time for each control and shutdown rod. The Stationary Gripper Release Time is a combination of a Trip Signal Delay Time (i.e. Delay between Reactor Trip Breaker opening and the trip signal to the RDTC) and the delay between power interruption (i.e. trip signal to the RDTC) and the rod's initiation of its free fall. The PAT evaluated the traces from the RDTC looking for a delay of each rod's initiation of free fall the RDTC's trip signal. The PAT also measured the "Trip Signal Delay Time" while in Mode 5.

This PAT also evaluated the traces from the rod drops in the area of the decelerating devices (i.e. dash pots) entry looking for proper performance of the dash pots.

5.6 Rod Drop Time Measurement and Stationary Gripper Release Timing (2-PAT-3.8) (continued)

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 Each CRDM unlatches upon opening the Reactor Trip Breakers.

Testing confirmed that each CRDM unlatched upon opening the Reactor Trip Breakers in Mode 5 and Mode 3.

3.2 The rod drop times for all shutdown and control rods, dropped from the fully withdrawn position, are within the limits specified in the Technical Specifications.

This Acceptance Criteria was successfully met by performance of 2-SI-85-10, Rod Drop Time Measurement Using Rod Drop Test Computer.

3.3 Rod drop time evaluations against the Two-Sigma statistical limits calculated resulted in either criteria below being applicable:

- Rod drop time was within the bounds of the lower and upper Two-Sigma statistical limits.

- Rods with rod drop times that fell outside of the bounds of the lower and upper Two-Sigma statistical limits have been dropped greater than or equal to 3 additional times. The results of the rod drop times were consistent (i.e. within a band of 50 milliseconds or less) and continue to meet Technical Specification criteria specified in 2-SI-85-10, Rod Drop Time Measurement Using CERPI Rod Drop Test Computer.

During the Mode 3 performance, Rods D-2 and M-14 failed to meet a 2 sigma statistical limit for the first rod drop. Three additional rod drops were successfully performed. (See CR 1169659)

3.4 The stationary gripper release time for all rods was <150 msec. This was the requirement in Unit 2 UFSAR Chapter 4.

During the Mode 5 performance the stationary gripper release time was conservatively determined to be 45 msec. Mode 3 release time was 50 msec.

5.6 Rod Drop Time Measurement and Stationary Gripper Release Timing (2-PAT-3.8) (continued)

3.5 The Trip Signal Delay Time was <100 msec.; as accounted for in 2-SI-85-10.

The Trip Signal Delay Time was determined to be 50 msec.

Review Criteria

None

4.0 Problems

[1] CR 1128964: During the Mode 5 performance the RDTC plots for each rod were inverted from the expected response. This issue did not impact the performance of this test and resolution of the CR occurred prior to the Mode 3 performance of this test.

[2] CR 1169659: Rods D-2 and M-14 did not meet a two-sigma statistical limit for the first rod drop in Mode 3. Three additional rod drops were performed and all Acceptance Criteria met.

5.7 Reactor Trip System (2-PAT-3.10)

This test was performed in Mode 5 as directed by 2-PAT-3.0, Post Core Loading Precritical Test Sequence. Testing was started on 1/13/16 and field work completed on 1/20/16.

1.0 <u>Test Objectives</u>

The objectives of this test were to:

1.1 Demonstrate proper functioning of the Reactor Trip System. This objective was accomplished by demonstrating that:

1.1.1 The reactor trip breakers can be opened manually from the Main Control Room (MCR)

1.2.2 Interlocks permit momentary closure of both reactor trip bypass breakers and then cause a reactor trip.

1.3.3 The reactor trip bypass breakers maintain the rod drive mechanisms energized when the associated reactor trip breaker is opened for test.

1.4.4 With one reactor trip bypass breaker closed, placing the opposite SSPS train channel in test causes both reactor trip breakers and the bypass breaker to open.

1.2 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 19, Reactor Trip System Test Summary.

2.0 <u>Test Methods</u>

2.1 Performance of this test (2-PAT-3.10) was completed with the unit in Mode 5 and RCS pressure greater than 100 psig to satisfy the requirements necessary for performance of 2-TRI-85-1, Reactivity Control Systems Movable Control Assemblies, as required by Technical Surveillance Requirement 3.1.7.

During performance of 2-TRI-85-1 an issue was encountered in which the rods controlled by the 2BD power cabinet (SB Group 2, CB Group 2 and, CD Group 2) did not respond to outward rod demand. Further details on this issue are documented in CR 1126661 found in the write-up for 2-PAT-3.1.

5.7 Reactor Trip System (2-PAT-3.10) (continued)

2.2 Section 6.2 verified both Main Control Room (MCR) Reactor Trip handswitches (2-RT-1 and 2-RT-2) generated a reactor trip and the associated indications appropriately.

2.3 Section 6.3 verified the electric interlocks prevented both bypass breakers from being closed simultaneously and resulted in a reactor trip due to a general warning in both trains of SSPS.

2.4. Sections 6.4 and 6.5 verified that when a reactor trip bypass breaker was closed placing the opposite SSPS train in test resulted in a reactor trip due to the generation of simultaneous general warnings in both trains.

2.5 Section 6.6 verified that the bypass breakers maintained the control rod drive mechanisms energized when the associated reactor trip breaker was opened due to injection of a simulated Reactor Protection System trip signal on the associated SSPS train.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 Reactor trip breakers (RTA and RTB) can be opened manually with hand switches 2-RT-1 and 2-RT-2.

Reactor trip breakers (RTA and RTB) were opened manually with both hand switches 2-RT-1 and 2-RT-2.

3.2 Electrical interlocks trip both reactor trip bypass breakers (BYA and BYB) when both bypass breakers are closed due to simultaneous general warning reactor trip signals being sent to the Reactor Trip Breakers (RTA and RTB)

Electrical interlocks successfully tripped both reactor trip bypass breakers when both bypass breakers were closed.

5.7 Reactor Trip System (2-PAT-3.10) (continued)

3.3 With one reactor trip bypass breaker (BYA or BYB) closed, placing the opposite SSPS train in test causes both reactor trip breakers (RTA and RTB) and the bypass breaker (BYA or BYB) to open due to simultaneous general warning reactor trip signals being sent to the reactor trip breakers (RTA and RTB)

Section 6.4 tested bypass breaker B in conjunction with SSPS Train A was completed successfully and all Acceptance Criteria were met as stated.

During performance of Section 6.5 a previously known and expected indication issue related to placing SSPS Train B Multiplexer test switch in INHIBIT, as originally documented in CR 1126043, was encountered. After verification that indications received were the same as those previously documented in CR 1126043 testing continued as the erroneous indications had no impact upon performance of 2-PAT-3.10 and Section 6.5 was completed satisfactorily with the exception of a procedure error which was identified in CR 1126802.

3.4 The reactor trip bypass breakers (BYA or BYB) maintain the rod drive mechanisms energized when the associated reactor trip breaker (RTA or RTB) is opened by injection of a simulated Reactor Protection System trip signal on the associated SSPS train

Each reactor trip bypass breaker maintained the rod drive mechanisms energized when the associated reactor trip breaker was opened. Section 6.6 was completed satisfactorily with the exception of a procedure error which was identified in CR 1126802.

<u>Review Criteria</u>

None

4.0 Problems

[1] CR 1126802 was generated because of procedure errors, which assumed breakers racked to the test position would still get the GEN WARNING alarm and MCR light indication for breaker position. It was determined that the procedure errors were minor and did not affect the test, including Acceptance Criteria Steps 6.5[14]D, 6.6[13]D, and 6.6[27]C that verified breaker position lights for bypass breakers not connected. After validation of the procedure error and its impacts, testing within 2-PAT-3.10 continued and Section 6.3 was completed satisfactorily.

Setup of Section 6.5 places the Reactor Trip Bypass Breaker B (BYB) in the test position; therefore, MCR indication lights for BYB are not illuminated. This section verifies that the reactor trips when Reactor Trip Bypass Breaker A (BYA) is closed and SSPS Train B is placed in Test; this was successfully performed with associated lights illuminated for BYA.

Section 6.6 verifies the Reactor Trip Bypass Breakers function to prevent a Reactor Trip during testing of Reactor Trip Breakers. During this section each of the Bypass Breakers were installed in the connected position (one at a time) with appropriate light indication.

5.8 Adjustment of Steam Flow Transmitters at Minimal Flow (2-PAT-3.11)

This test was performed with the plant in Mode 3 at normal operating pressure and temperature, as specified in 2-PAT-3.0, Post Core Loading Pre-Critical Test Sequence. The test was started on 5/6/16 and field work complete on 5/7/16.

1.0 <u>Test Objectives</u>

The objectives of this test were to:

1.1 Verify/adjust the output of the eight steam flow transmitters for "zero" output with minimal steam flow.

1.2 Satisfy the Mode 3 objective in the UFSAR Table 14.2-2, Sheet 21, Calibration Of Steam And Feedwater Flow Instrumentation At Power Test Summary.

2.0 <u>Test Methods</u>

The plant was in Mode 3 at normal operating temperature and normal operating pressure. Steam flow was reduced to minimal by shutting a MSIV, one loop at a time. With the MSIV closed, each steam flow transmitter on the associated main steam line was verified/adjusted for a "zero" output. This was repeated for each main steam line.

3.0 <u>Test Results</u>

All Acceptance/Review Criteria were met or resolved as delineated below.

<u>Acceptance Criteria</u>

None

<u>Review Criteria</u>

3.1 At minimum steam flow, the output from each steam flow transmitter and its associated loop reflects zero flow as demonstrated by the following:

A. D/P Test Point: 0.19829 Vdc (0.19641 to 0.20016 Vdc)
B. Flow Test Point: 0.2000 Vdc (0.1972 to 0.2028 Vdc)
C. Computer Test Point: 0.2000 Vdc (0.1972 to 0.2028 Vdc)
D. Computer Point: 0.0 KBH (-275 to 275 KBH)

5.8 Adjustment of Steam Flow Transmitters at Minimal Flow (2-PAT-3.11) (continued)

The data below was collected and the output flow was verified/adjusted within the Review Criteria requirements.

Transmitter	D/P Test Point (Vdc)	Flow Test Point (Vdc)	Computer Test Point (Vdc)	Computer Point (KBH)
2-FT-1-3A*	0.19810	0.19983	0.20103	4
2-FT-1-3B	0.19685	0.20055	0.20075	4
2-FT-1-10A	0.19782	0.20038	0.20027	1
2-FT-1-10B*	0.19780	0.20071	0.20121	6
2-FT-1-21A*	0.19867	0.20160	0.20013	1
2-FT-1-21B	0.19960	0.20038	0.20055	6
2-FT-1-28A	0.19811	0.20070	0.20050	2
2-FT-1-28B	0.19690	0.20007	0.20051	3

* adjustment made

4.0 Problems

There were no significant problems encountered during the performance of this test.

5.9 Control Rod Drive Mechanism Timing (2-PET-106)

This test was performed as part of test sequence 2-PAT-3.0, Post Core Loading Precritical Test Sequence. The test began via WO 117705850 on 04/13/16 and was field work completed on 5/6/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Verify the acceptability of the Control Rod Drive Mechanism, (CRDM), current order timing, current order amplitudes, and rod withdrawal speed.

1.2 Partially satisfy the requirements of UFSAR Table 14.2-2, Sheet 7, Control Rod Drive Mechanism Timing Test Summary.

2.0 Test Methods

The test was required to be performed following fuel loading. Since the CRDM latch assembly must be submerged in water for proper operation, a minimum RCS pressure of 100 psig was required. The test was run at nominal hot plant conditions. Reactor Engineering verification of current boron concentration being adequate to perform this test by being equal to or greater than the refueling boron concentration was required.

With the reactor trip breakers closed and the lift coils verified to be connected, a selected bank was withdrawn and then reinserted approximately 10 steps to obtain the CRDM readings.

The test objectives were accomplished by monitoring the CRDM coil current profiles to verify that the stationary gripper, movable gripper, and lift coil current order changes occur at the proper time during the 780 msec. rod stepping cycle; that stationary, movable and lift coil currents are properly regulated to full current values within acceptable ranges during rod withdrawal and insertion operations; that shutdown bank rod withdrawal speed is a nominal 64 steps/min and control bank rod withdrawal speed is a nominal 48 steps/min.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 Current Order Timing

The times at which the lift, movable, and stationary current orders change, after the start of rod motion, are within 10 msec. of the expected times during rod withdrawal and insertion operations.

5.9 **Control Rod Drive Mechanism Timing (2-PET-106)** (continued)

The lift, movable, and stationary current orders for all CRDMs were within 10 msec. of the expected times during rod withdrawal and insertion operations.

3.2 Coil Current Amplitudes

Stationary, movable and lift currents are regulated by circuity internal to each power cabinet. The reduced and full current nominal values are not critical, cannot be adjusted, but could be an indication of a regulation failure. Measured Values outside the nominal ranges below should be evaluated and documented by the system engineer.

3.2.1 Lift Coil - full	35 to 47.2 amperes (equivalent to 438 to 590 mVdc measured across a 0.0125 ohm resistor)
3.2.2 Lift coil - reduced	13 to 19.7 amperes (equivalent to 163 to 246mVdc measured across a 0.0125 ohm resistor)
3.2.3 Movable Gripper Coil - full	7 to 9.2 amperes (equivalent to 438 to 575 mVdc measured across a 0.0625 ohm resistor)
3.2.4 Stationary Gripper Coil - full	7 to 9.2 amperes (equivalent to 438 to 575 mVdc measured across a 0.0625 ohm resistor)
3.2.5 Stationary Gripper coil - reduced	3.8 to 4.8 amperes (equivalent to 238 to 300 mVdc measured across a 0.0625 ohm resistor)

All stationary, movable, and lift currents amplitudes for all CRDMs were within the Acceptance Criteria with the exception of D08, B10, F14, F10, and D12 lift coil reduced currents which were greater than the 19.7 amperes criteria. This issue was previously evaluated in CR 1128950.These reduced lift currents were all less than 21 amperes which was evaluated by Westinghouse as acceptable in Westinghouse Letter WBT-D-5604.

5.9 Control Rod Drive Mechanism Timing (2-PET-106) (continued)

3.3 Rod Withdrawal Speed

3.3.1 Shutdown Bank withdrawal speed nominal 64 steps per minute.

The measured Shutdown Bank withdrawal speed was approximately 64 steps per minute and did not exceed the nominal value.

3.3.2 Control Bank withdrawal speed nominal 48 steps per minute.

The measured Control Bank withdrawal speed was approximately 48 steps per minute and did not exceed the nominal value.

Review Criteria

None

4.0 Problems

There were no significant problems encountered during the performance of this test.

6.0 INITIAL CRITICALITY AND LOW POWER PHYSICS TESTING

6.1 Initial Criticality and Low Power Test Sequence (2-PAT-4.0)

2-PAT-4.0 started with prerequisites on 5/12/16 and completed on 5/24/16.

1.0 Test Objectives

The objective of this test was to:

1.1 Provide governance of the sequence of the Power Ascension Testing in Mode 2.

The following PATs/PETs were sequenced for performance by 2-PAT-4.0:

- 2-PET-201 * Initial Criticality and Low Power Physics Testing
- 2-PET-103 Reactivity Computer (ADRC)
- 2-PET-304 * Operational Alignment of NIS
- 2-PAT-1.5 * Loose Parts Monitoring System
- 2-PAT-1.10 * Integrated Computer System (ICS)
- RCI-159 * Radiation Baseline Surveys

Note: * Indicates that the test is performed at multiple test plateaus. The description of the testing is documented in the section (plateau) in which it was completed.

2.0 Test Methods

Prerequisite actions for this Power Ascension Test (PAT) started on 5/12/16 and completed on 5/22/16 and included verification of the following major items:

- Preoperational tests completed to allow entry into Mode 2
- TVA-SPP-30.010, Initial Synchronization of TVA Generating Assets to TVA's Transmission System notification
- Reactivity Control Plans are developed to support testing
- 2-PET-201, Initial Criticality and Low Power Physics Testing, has been initiated
- Section 4.0 of 2-PET-103, Reactivity Computer (ADRC), has been performed

Prior to initiation of the performance section, a cool down was initiated on 5/16/16 to 360°F to replace a failed hot leg RTD. RCS temperature was stabilized between 355°F and 365°F at 22:59 on 5/16/16. The unit was placed in Mode 4 on 5/18/16 at 23:58 to facilitate SSPS testing. After completion of testing Unit 2 re-entered Mode 3 at 04:15 on 5/20/16 and began a heat up to normal operating temperature and pressure. NOTP was reached on 5/21/16 at 01:00. On 5/23/16 at 01:04 the unit entered Mode 2.

The reactor was taken critical on 5/23/16 at 02:16.

2-PAT-4.0 governed initial criticality and the low power testing greater than 3 percent and less than 5 percent reactor power. Applicable portions of the following procedures were initiated and completed as appropriate.

- 2-PET-201, Initial Criticality and Low Power Physics Testing - Completed 5/23/16 with all criteria met.
- 2-PET-103, Reactivity Computer (ADRC), completed 5/23/16 with all criteria met.
- 2-PET-304, Operational Alignment of NIS, completed 5/23/16 with all criteria met.
- 2-PAT-1.5, Loose Parts Monitoring System, was completed on 5/24/16 with all criteria met. CR 1171424 documents three channels removed from service.
- 2-PAT-1.10, Integrated Computer System (ICS), completed 5/24/16 CR 1174334 documents exceeding the MED between T0457A and MCR indicator 2-TI-62-29, RCP 3 LWR RADIAL BRG Temp.
- RCI-159, Radiation Baseline Surveys - completed 5/31/16. No Acceptance or Review Criteria were associated with this procedure.

Details of the performance of each PAT procedure is contained in the individual summaries of the associated procedures.

3.0 Test Results

All Acceptance/Review Criteria were contained within the tests sequenced by this test.

4.0 Problems

Problems encountered are addressed in the following discussions of each test sequenced by 2-PAT-4.0.

6.2 Reactivity Computer (ADRC) (2-PET-103)

This test was performed as part of test sequence 2-PAT-4.0, Initial Criticality and Low Power Test Sequence. Field performance of 2-PET-103 was commenced on 05/15/16. The purpose of this procedure is to ensure that the Advanced Digital Reactivity Computer (ADRC) is capable of reactivity measurements in support of Low Power Physics Testing (LPPT) per 2-PET-201. This procedure was completed on 05/23/16 following completion of LPPT.

1.0 Test Objectives

The objectives of this test were to:

1.1 Perform installation of the ADRC

1.2 Perform the calibration and setup of the ADRC prior to reactivity measurements.

1.3 Provide instructions for connecting/restoring the RCS Temperature and Rods Move signals to/from the ADRC

2.0 Test Methods

This test provided instructions for setup and installation of the ADRC for LPPT. This test connected a RCS T_{avg} signal from the Unit 2 Auxiliary Instrument Room to the ADRC, connected the Power Range detectors Top and Bottom signals and the "Rods Move" signal to the ADRC, and provided instructions on initial checkout of the reactivity computer.

Proper installation was verified by performing the initial checkout and initial exponential test. The initial checkout ensured that the ADRC was loaded with the correct constants and reactivity data consistent with WBN Unit 2 Cycle 1 core design. After input data was confirmed, the initial exponential test was conducted using a simulated signal for reactor flux. This calculated reactivity was verified to be within 1.0% of the theoretical value. This test ensured that the ADRC was correctly calculating reactivity with appropriate input data.

Once physics testing was complete, steps were given to remove all installed cables and return the plant to its original state.

6.2 Reactivity Computer (ADRC) (2-PET-103) (continued)

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1. The absolute value of the PREDICTED vs MEASURED error, the percent difference between the ADRC "predicted" reactivity and the "measured" reactivity is < 1.0% during the ADRC Internal Exponential Test.

The difference between the "predicted" reactivity and "measured" reactivity was found to be ~0.03%, within the 1.0% criteria specified by the procedure.

Review Criteria

None

4.0 Problems

The following issues were encountered during Reactivity Computer setup per 2-PET-103:

[1] While verifying the inputs to the ADRC were correct, it was noted that the value for the prompt neutron lifetime was inconsistent between the value stated in the eNuPOP compared to the value being used by the ADRC. The eNuPOP listed a value of 19.718 microseconds while the ADRC was found to have a value of 19.716 seconds. Following consultation with the fuel vendor, it was determined that both values were acceptable (per Westinghouse letter NF-TV-16-24) and showed the small difference due to being calculated by two separate versions of code. The value listed in the ADRC was calculated using a later version of the ANC code. A one-time-only change was generated for this procedure to allow for this difference. The procedure originally stated that the values had to be "identical." The one-time-only change allowed for the values to be "consistent."

6.3 Initial Criticality and Low Power Physics Testing (2-PET-201)

The test was performed as part of test sequence 2-PAT-4.0, Initial Criticality and Low Power Test Sequence. Field performance of 2-PET-201 was commenced on 5/22/16 and initial criticality was achieved at 02:16 on 5/23/2016. The test was completed on 05/23/16 with successful completion of initial criticality, rod worth measurements (using Dynamic Rod Worth Measurement (DRWM) method), boron endpoint measurements, and isothermal temperature coefficient testing.

1.0 <u>Test Objectives</u>

The objectives of this test were to:

1.1 Dilute the reactor to criticality in a cautious and controlled manner

1.2 Perform Mode 2 Low Power Physics Testing in a cautious and controlled manner, including:

1.2.1 Measuring the integral worth of the control and shutdown rod banks.

1.2.2 Measuring the ARO critical boron concentration.

1.2.3 Measuring the ARO ITC.

1.3 This test and associated SIs satisfied the requirements of UFSAR Table 14.2-2:
Sheet 22, Initial Criticality Test Summary.
Sheet 23, Determination Of Core Power Range For Physics Testing Test Summary.
Sheet 24, Moderator Temperature Coefficient Test Summary. (2-SI-0-23, Moderator Temperature Coefficient Determination at BOL)
Sheet 25, Rod And Boron Worth Measurements Test Summary.
Sheet 26, Core Reactivity Balance, Acceptance Criteria 1. (2-SI-0-12, Core Reactivity)

Note: Sheet 26, Core Reactivity Balance, Acceptance Criteria 2 is documented in 2-SI-0-12, Core Reactivity, at full power.

2.0 Test Methods

Initial reactor startup was conducted via dilution to critical while all shutdown and control banks were fully withdrawn. The dilution began at 65 gpm and the reactor was monitored by use of ICRR. See Figure 6.3-1, ICRR vs Primary Water (N31, N32)
When the ICRR reached 0.3, the dilution was terminated. After criticality was achieved and power increased, control rods were inserted to zero the startup rate with reactor power near $1x10^{-3}$ % power.

With the reactor stable, a "bite check" was then performed to determine if the inserted worth of Control Bank D was between 40 to 75 pcm. An RCS boration was performed to establish an inserted worth of 62 pcm. A reactor exponential test was then conducted while finding the point of adding heat to set the physics testing range.

With the Physics Testing Range met, the DRWM testing began by withdrawing CBD in Manual to the full out position. Once flux reached the appropriate level on the reactivity computer CBD was inserted continuously in individual bank select until 0-5 steps withdrawn. When data collection was complete, CBD was restored to the full out position. This process, of measuring rod bank worth, was repeated for each remaining control and shutdown bank, in individual bank select. The reactor was then brought back to a stable condition in Manual with Control Bank D slightly inserted.

The boron endpoint was then calculated using the measured bank worth data by use of the ADRC. This information is used to determine the ARO HZP No XE critical boron concentration. The ITC was then measured by initiating a constant rate cooldown, at less than 30 deg F/hr. When data collection was complete, a constant rate heatup, also at less than 30 deg F/hr, was initiated. Both sets of data were analyzed to calculate an average ITC and converted to a MTC, accounting for the Doppler reactivity coefficient.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

6.3 Initial Criticality and Low Power Physics Testing (2-PET-201) (continued)

Acceptance Criteria

3.1 Advanced Digital Reactivity Computer (ADRC) Checkout

The indicated reactivity is within ±4% or ± 1 pcm of the theoretical reactivity for each reactor exponential measurement.

Indicated reactivity during the reactor exponential test for ADRC checkout was measured at 25.0 pcm with a predicted reactivity of 24.8 pcm. This resulted in a difference of 0.94% or 0.2 pcm difference.

3.2 Control and Shutdown Bank Worths (DRWM criteria)

The sum of the measured bank worths is greater than or equal to (100%-RWU) times the sum of the predicted bank worths.

The RWU, Rod Worth Uncertainty, is given as 10% for Unit 2 Cycle 1. The sum of the measured bank worths was measured to be 1.2% greater than the predicted bank worths. This value is greater than 90% (100%-RWU) of the predicted bank worths.

3.3 Boron Endpoint Measurement

Boron endpoint Acceptance Criteria is verified in 2-SI-0-12, Core Reactivity. (2-PET-201 verified that 2-SI-0-12 was successfully completed). The Technical Specification Acceptance Criteria within 2-SI-0-12 is for measured Mode 2 HZP ARO critical boron concentration shall be within the reactivity equivalence of ±1000 pcm of the predicted HZP ARO critical boron concentration

The Boron Endpoint Acceptance Criteria was met via performance of 2-SI-0-12 (WO 117827845) following data collection from 2-PET-201. The Boron Endpoint was measured at 1089 ppm. The predicted value was 1034 ppm. This resulted in a difference of 55 ppm, or -569.9 pcm.

3.4 Temperature Coefficient

The Moderator Temperature Coefficient (MTC) Acceptance Criteria is verified in 2-SI-0-23, Moderator Temperature Coefficient Determination at BOL (WO 115947713).
(2-PET-201 verified that 2-SI-0-23 was successfully completed.) The Technical Specification Acceptance Criteria within 2-SI-0-23 are:

3.4.1 The MTC is less than or equal to 0.0 pcm/°F at HZP.

6.3 Initial Criticality and Low Power Physics Testing (2-PET-201) (continued)

3.4.2 The MTC is less than or equal to the Beginning of Cycle MTC as-measured criterion specified in the COLR.

Both MTC Acceptance Criteria were met by successful performance of 2-SI-0-23. The MTC was measured by 2-PET-201 to be -3.515 pcm/°F, which is less than the 0.0 pcm/°F limit and below the COLR limit of -3.33 pcm/°F.

3.5 Zero Power Physics Testing Range

The zero power physics testing range is determined such that reactivity feedback from nuclear heating does not compromise the measurements.

The zero power physics testing range was determined to not have any reactivity feedback affects prior to performing rod worth, Boron Endpoint or ITC testing.

Review Criteria

The Review Criteria are listed below with two noted failures.

3.6 ADRC Checkout

3.6.1 The indicated reactivity is within $\pm$2% or $\pm$1 pcm of the theoretical reactivity for each measurement.

Indicated reactivity during the reactor exponential test for ADRC checkout was measured at 25.0 pcm with a predicted reactivity of 24.8 pcm. This resulted in a difference of 0.94% or 0.2 pcm difference.

3.6.2 The reactivity traces do not exhibit excessive noise level ($\pm$2 pcm).

During the determination of the physics testing range, reactivity traces were reviewed and confirmed to not exhibit excessive noise outside of the specified Review Criteria tolerance.

3.6.3 The reactivity indication is stable as a function of flux level (no obvious dependence on the flux input level).

During the determination of the physics testing range, reactivity traces were reviewed and confirmed stable indication of reactivity as a function of flux level with no obvious dependence of the flux input level.

6.3 Initial Criticality and Low Power Physics Testing (2-PET-201) (continued)

3.7 Rod Worth Measurement (DRWM Criteria)

3.7.1 The measured worth of all banks are within ±10% or ±75 pcm of the prediction, whichever is greater.

All shutdown and control banks were within the ±10% and ±75 pcm of the predicted values.

3.7.2 The sum of the measured worths of all banks are within +(0.8*RWU)% of the prediction.

Bank	Measured (pcm)	Predicted (pcm)	M-P (pcm)	100*(M/P-1) (%)
CD	1,304.2	1,338.5	-34.3	-2.6%
CC	1,061.3	1,052.7	8.6	0.8%
CB	794.4	743.2	51.2	6.9%
CA	910.0	951.4	-41.4	-4.4%
SD	437.7	434.4	3.3	0.8%
SC	447.4	434.4	13.0	3.0%
SB	1,055.7	1,017.1	38.6	3.8%
SA	424.1	389.4	34.7	8.9%
Total	6,434.8	6,361.1	73.7	1.2%

All shutdown and control banks were within ±10% and ±75 pcm of the predicted values. The sum of the measured worths of all banks were within ±(0.8*RWU)% of the prediction.

3.8 Boron Endpoint

3.8.1 Measured ARO boron endpoint is within ±50 ppm of the predicted boron endpoint.

The measured ARO boron endpoint was measured as 1089 ppm, which was 55 ppm higher than the predicted boron endpoint of 1034 ppm. CR 1173995 initiated to document Review Criteria failure.

3.8.2 Measured ARO boron endpoint is within ±500 pcm equivalent boron.

The measured ARO boron endpoint was measured as -569.9 pcm different from predicted values.
CR 1173995 captures this failed Review Criteria also.

3.9 Temperature Coefficient

The Measured ITC is within ± 2 pcm/°F of the predicted ITC.

The ITC was measured as -5.305 pcm/°F with a predicted value of -6.67 pcm/°F.

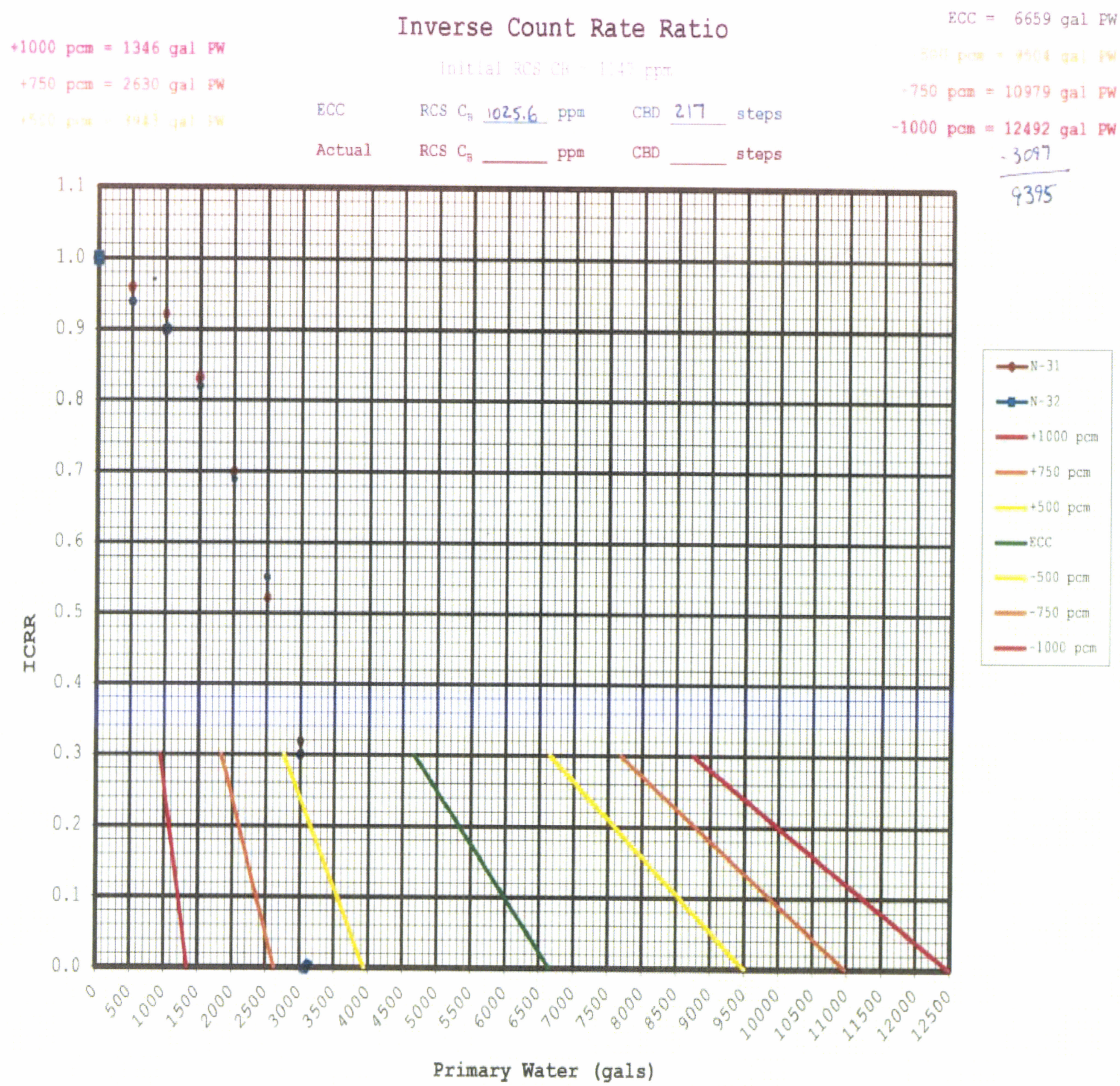
4.0 Problems

[1] CR 1173995: Both Review Criteria for Boron Endpoint results were not met. The Boron Endpoint was measured to be -569.9 pcm or 55 ppm from predicted value. The Acceptance Criteria were met for reactivity balance.

Reactor Engineering, Nuclear Fuel and Westinghouse concluded that there were no safety concerns or issues resulting from this difference.

FIGURE 6.3-1

ICRR vs Primary Water (N31, N32)



7.0 POWER ASCENSION TESTING

7.1 Test Sequence for 30% Plateau (2-PAT-5.0)

This test started on 5/17/16 and was completed on 6/16/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Define the plant operational requirements in conjunction with 2-GO-3, Unit Startup from Less than 4% Reactor Power to 30% Reactor Power.

1.2 Ensure those requirements were met in order to permit power escalation from Mode 2 conditions with reactor power ≤ 5% Rated Thermal Power (RTP) to 30%.

1.3 Specify the order of test performance at the 30% plateau.

The following PATs/PETs were sequenced for performance by 2-PAT-5.0:

- 2-PAT-1.4 * Pipe Vibration Monitoring
- 2-PAT-1.5 * Loose Parts Monitoring System
- 2-PAT-1.6 * Startup Adjustments of Reactor Control System
- 2-PAT-1.7 * Operational Alignment of Process Temperature Instrumentation
- 2-PAT-1.8 * Thermal Expansion of Piping Systems
- 2-PAT-1.10* Integrated Computer System (ICS)
- 2-PAT-1.11* RVLIS Performance Test
- 2-PAT-1.12* Common Q Post Accident Monitoring System
- 2-PAT-5.1 * Dynamic Automatic Steam Dump Control
- 2-PAT-5.3 Automatic Steam Generator Level Control, Transients at Low Power
- 2-PAT-5.4 Calibration of Steam and Feedwater Flow Instruments at 30% Power
- 2-PET-301 * Core Power Distribution Factors
- 2-PET-304 * Operational Alignment of NIS
- RCI-159 * Radiation Baseline Survey

Note: * Indicates that the test is performed at multiple test plateaus. The description of the testing is documented in the section (plateau) in which it was completed.

This test and WO 116916855 for (WINCISE) Post-Critical System Calibration (WNA-TP-04724-WBT) satisfy UFSAR Table 14.2-2, Sheet 12, Incore Instrumentation System Test Summary, Acceptance Criteria 2.

2.0 Test Methods

Prerequisite actions for this Power Ascension Test (PAT) started on 5/17/16 and completed on 5/25/16 and included verification of the following major items:

- 2-PAT-4.0, Initial Criticality and Low Power Test Sequence, has been completed.
- NPG-SPP-10.4, Reactivity Management Program, Reactivity Control Plans were developed to support the planned testing for this sequence.
- WO 116916855 implemented vendor procedure WNA-TP-04724-WBT, Westinghouse Incore Information Surveillance & Engineering (WINCISE) Post Critical System Calibration.
- WINCISE incore signal quality verification was in progress by implementation of applicable section of vendor procedure WNA-TP-04724-WBT.
- Reactor power was ≤ 5% RTP
- RCS pressure was between 2220 to 2250 psig
- Section 6.3 of 2-PET-304, Operational Alignment of NIS, to adjust the Power Range High Flux Level Trip setpoints for testing at the 30% Plateau was complete.

On 5/25/16 the performance section of 2-PAT-5.0 was begun and a power increase to 6-9 percent was initiated. Mode 1, ≥ 5% power, was reached at 03:33 on 5/25/16.

2-PAT-5.3, Automatic Steam Generator Level Control Transients at Low Power, Section 6.1, was completed on 5/26/16 with all criteria met.

RCI-159, Radiation Baseline Surveys, was completed on 5/31/16. No Acceptance or Review Criteria were associated with this procedure.

2-PAT-5.1, Dynamic Automatic Steam Dump Control, Sections 6.3, 6.4, and 6.5 were started on 5/25/16 and completed on 5/27/16. Section 6.3 and 6.4 were completed with all criteria met. Section 6.5 was completed after an Urgent Change to the procedure was approved by the TRG to change the load rejection testing criteria per Westinghouse Letter LTR-SCS-16-23 (LTR-PCSA-16-23). The revised Acceptance Criteria was met for Section 6.5.

On 5/27/16, with reactor power between 13 and 14 percent, the turbine was rolled and subsequently stopped due to noise in the area of the turning gear. The turbine was shut down and subsequently re-rolled and the noise repeated at approximately 400 rpm. The turbine was shutdown and the decision made to place the Unit in Mode 3 for turbine repairs.

7.1 Test Sequence for 30% Plateau (2-PAT-5.0) (continued)

On 5/28/16 at 01:54 the Unit re-entered Mode 3 after a manual reactor trip. The generator was purged and a clearance placed on the turbine for inspection. On 5/31/16 Unit 2 entered Mode 2 at 12:00 followed by taking the reactor critical at 13:39. Mode 1 entry was made on 17:49 on 5/31/16. Unit 2 was synchronized to the grid on 6/3/16 at 20:39 and power increased to 15 percent. As power increased a steam leak required a manual turbine trip on 6/4/16 at 16:58. On 6/5/16 at 11:40 the turbine was again tied to the grid and at 12:27 a Reactor Trip - Safety Injection occurred due to the #1 governor valve failing to the open position.

After repairs to the governor valve, as well as additional work on 2B Main Feed Pump, the unit was returned to Mode 2 on 6/8/16 at 01:39. Mode 1 was re-entered 6/8/16 at 09:32. On 6/9/16 at 06:40 the generator was synchronized to the grid. An un-isolable steam leak required a turbine trip on 6/9/16 at 17:52. Repairs were made and Unit 2 was synchronized to the grid at 13:23 on 6/11/16. Power was increased to allow testing between 25 and 30 percent with the following Power Ascension Test being completed as scheduled:

- 2-PAT-1.4, Pipe Vibration Monitoring, completed on 6/15/16 with all criteria met for observations at the 30% Plateau.
- 2-PAT-5.3, Automatic Steam Generator Level Control Transients at Low Power was completed on 6/15/16 with all Acceptance Criteria met. CR 1181278 was initiated to document one Review Criteria that was not met. An engineering evaluation determined this did not affect the performance of the test nor invalidate any of the test results and testing should proceed to the next plateau.
- 2-PAT-1.5, Loose Parts Monitoring System, was completed on 6/13/16 with all criteria met. CR 1171424 documents three channels removed from service.
- 2-PAT-1.8, Thermal Expansion of Piping Systems, was field work complete on 6/15/16 with no issues noted.
- 2-PAT-1.10, Integrated Computer System (ICS), was completed on 6/14/16 with all criteria met. CR 1181784 was written to address a database error but did not affect this plateau performance.
- 2-PAT-1.11, RVLIS Performance Test, applicable sections were completed on 6/13/16 with all criteria met.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, applicable sections were completed on 6/13/16 with all criteria met.

- 2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, was completed on 6/15/16. All Acceptance Criteria were met. Two Review Criteria concerning parameters related to Delta T failed. The OTDT calculated by Eagle-21 and provided by the MMI carts indicated approximately 158% and the MCR indicators maximum value is 150%. It was expected the reading from Eagle-21 was accurate and the MCR meters were ranged such that they cannot read the higher value. Additional data was taken at higher power ranges and the meters came on scale with no issue. CR 118246 was written.
- 2-PAT-5.4 Calibration of Steam and Feedwater Flow Instruments at 30% Power was completed on 6/15/16 with all criteria met for the 30% Plateau.
- 2-PAT-1.6 Startup Adjustments of Reactor Control System was completed on 6/15/16. This was data taking only with no Review or Acceptance Criteria at this plateau.

Additionally, Engineering completed the following procedures, with no issues, to support their testing at the 27-29 percent power level:

- 2-TI-41 - Incore Flux Mapping
- 2-TRI-0-22 - PDMS Operability
- 2-SI-0-21 - Excore QPTR & Axial Flux Difference
- 2-SI-92-3 - Incore-Excore Cross Calibration Data
- 2-TI-7.020 - PDMS Calibration
- 2-TI-6 - Calorimetric Calibration

After completion of all testing in this PAT it was noted that tempering flow isolations occurred that did not meet the requirements of Westinghouse letter WAT-D-6432. CR 1182320 was written to document tempering flow isolations that occurred as part of testing at this plateau. Corrective actions from this CR evaluated the length of the isolation and revised 2-SOI-2&3.01 adding a Precaution about WAT-D-6432.

3.0 Test Results

All Acceptance/Review Criteria were contained within the tests sequenced by this test.

4.0 Problems

Problems encountered are addressed in the following discussions of each test sequenced by 2-PAT-5.0.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1)

This test was performed as part of test sequences 2-PAT-3.0, Post Core Loading Precritical Test Sequence, and 2-PAT-5.0, Test Sequence for 30% Plateau. The test began on 1/8/16 and was field work completed on 5/27/16.

The steam dump valves were tested without steam flow during sequence 2-PAT-3.0 in accordance with Sections 6.1 and 6.2.

The steam dump valves were tested with steam flow during sequence 2-PAT-5.0 in accordance with Sections 6.3, 6.4 and 6.5.
The plant was less than 15% power, in Mode 1 with the main turbine not synchronized to the grid.

For Sections 6.6, the steam dump valves were tested for the deferral from Startup with steam flow during Mode 3. This was done to confirm stroke times in all three of the following simulated scenarios: modulate open, trip close, and trip open. Additionally, vibration testing on the valves which was deferred from Startup Testing was performed.

1.0 Test Objectives

The objectives of this test were to:

1.1 Verify the operation of the Steam Dump Control System. The Steam Dump Control System has three modes of control; Steam Pressure, Plant Trip, and Load Rejection. Each mode of control was tested to demonstrate stability following a small transient.

1.2 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 28, Dynamic Automatic Steam Dump Control Test Summary.

1.3 Address additional scope of testing added to the Power Ascension Testing Program for the deferred Turbine Bypass System (Condenser Steam Dump Valves) testing. This test was to verify the (12) Steam Dump Valves stroke times are within acceptable limits and to obtain vibration data on deferred Steam Dump valves not obtained during Hot Functional Testing in accordance with 2-PAT-1.4.

1.4 Satisfy the requirements of UFSAR Chapter 14, Table 14.2-1, Sheet 62, Main Steam System Test Summary by collection vibration data on deferred Steam Dump valves not obtained during Hot Functional Testing in accordance with 2-PAT-1.4.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

2.0 Test Methods

The steam dump control system is designed to maintain RCS average temperature by dumping steam to the condenser. This instruction functionally tested all three control modes (steam pressure control, plant trip control, and load reduction control) while reactor power was low (i.e., <15% power).

The functional test included modulating the valves open and closed, and tripping open all steam dump control valves using simulated signals while steam flow was isolated. The Steam Pressure controller was tested by varying reactor power and observing the controller automatically maintained steam header pressure by changing steam flow to the condenser. The Plant Trip controller was tested by simulating a reactor trip, varying reactor power, and observing controller parameters and output. The Load Rejection controller was tested by simulating the loss of load permissive, and observing controller parameters and output.

2-PAT-1.4, Pipe Vibration Monitoring, data was collected during the performance of the PAT.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

<u>Acceptance Criteria</u>

Note: There were no Acceptance Criteria for Sections 6.1 and 6.2.

3.1 Section 6.3 (Steam Pressure Controller)

3.1.1 After varying reactor power, the steam pressure controller maintains steam header pressure stable, as demonstrated by neither the steam header pressure signal nor the steam dump demand signal showing divergent oscillations.

During the transient neither the steam header pressure signal nor the steam dump signal showed a divergent oscillation.

3.1.2 After varying reactor power, steam pressure controller maintains steam header pressure stable, as demonstrated by the steam dump control system remaining in automatic throughout the transient.

The steam pressure controller maintained steam header pressure stable and the steam dump control system remained in automatic throughout the transient.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

3.2 Section 6.4 (Plant Trip Controller)

3.2.1 After varying reactor power, the plant trip controller maintains a stable Tavg as demonstrated by neither the RCS Tavg signal nor the steam dump demand signal showing divergent oscillations.

During the transient neither the RCS Tavg signal nor the steam dump demand signal showed a divergent oscillation.

3.2.2 After varying reactor power, the plant trip controller maintains a stable Tavg as demonstrated by the steam dump control system remaining in automatic without divergent oscillations.

The plant trip controller maintained Tavg stable and the steam dump control system remained in Automatic throughout the transient without divergent oscillations.

3.3 Section 6.5 (Load Rejection Controller)

3.3.1 The loss of load controller responds properly for the plant input signals to the controller.

The loss of load controller responded properly for the plant input signals to the controller.

3.4 Section 6.6 (Condenser Steam Dump valves stroke times)

3.4.1 Condenser steam dump valves modulate open, trip closed and trip open stroke times are within acceptable limits.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

Modulation times and local/remote stroke times indicated below are within Acceptance Criteria.

VALVE	MODULATE OPEN ≤ 20 sec	TRIP CLOSE ≤ 5 sec	TRIP OPEN ≤ 3 sec
2-FCV-1-103	6.5 / 5.71	3.0 / 2.7	2.2 / 1.76
2-FCV-1-104	6.2 / 4.61	2.25 / 2.90	2.32 / 2.12
2-FCV-1-105	13.31 / 9.69	3.20 / 3.10	2.80 / 2.54
2-FCV-1-106	9.34 / 9.29	2.96 / 3.11	2.60 / 2.08
2-FCV-1-107	5.40 / 6.20	2.74 / 2.50	2.80 / 2.33
2-FCV-1-108	9.11 / 5.89	2.80 / 2.88	2.82 / 2.35
2-FCV-1-109	9.53 / 6.9	3.41 / 3.36	2.94 / 2.08
2-FCV-1-110	9.16 / 7.88	3.48 / 2.68	2.62 / 2.18
2-FCV-1-111	4.02 / 3.46	3.08 / 3.18	2.56 / 1.62
2-FCV-1-112	4.67 / 4.26	2.76 / 2.67	2.19 / 1.68
2-FCV-1-113	6.39 / 6.0	3.18 / 3.10	2.00 / 2.10
2-FCV-1-114	9.96 / 9.41	3.00 / 3.50	2.95 / 2.30

Review Criteria

3.5 Section 6.1 (Static Valve Timing - Modulation)

3.5.1 The full open stroke length for each steam dump control valve is 2 3/4 inches to 3 inches.

Review criteria were met as delineated below:

Steam Dump Valve	Expected Stroke Length	Measured Stroke Length
2-FCV-1-103	2 3/4 - 3 in.	2.812
2-FCV-1-104	2 3/4 - 3 in.	2.75
2-FCV-1-105	2 3/4 - 3 in.	2.875
2-FCV-1-106	2 3/4 - 3 in.	2.875
2-FCV-1-107	2 3/4 - 3 in.	2.922
2-FCV-1-108	2 3/4 - 3 in.	2.81
2-FCV-1-109	2 3/4 - 3 in.	2.875
2-FCV-1-110	2 3/4 - 3 in.	2.812
2-FCV-1-111	2 3/4 - 3 in.	2.812
2-FCV-1-112	2 3/4 - 3 in.	2.75
2-FCV-1-113	2 3/4 - 3 in.	2.812
2-FCV-1-114	2 3/4 - 3 in.	2.812

7.1.1 **Dynamic Automatic Steam Dump Control (2-PAT-5.1)** (continued)

3.5.2 The opening modulation time for each Steam Dump Control Valve is less than 20 seconds upon the receipt of a 5% to 95% control signal step change.

Review criteria were met as delineated below:

Steam Dump Valve	Open Stroke Timing Requirement	Actual Open Stroke Timing
2-FCV-1-103	≤ 20 Seconds	2.76
2-FCV-1-104	≤ 20 Seconds	4.81
2-FCV-1-105	≤ 20 Seconds	10.47
2-FCV-1-106	≤ 20 Seconds	7.53
2-FCV-1-107	≤ 20 Seconds	7.10
2-FCV-1-108	≤ 20 Seconds	10.91
2-FCV-1-109	≤ 20 Seconds	6.91
2-FCV-1-110	≤ 20 Seconds	5.91
2-FCV-1-111	≤ 20 Seconds	6.72
2-FCV-1-112	≤ 20 Seconds	3.96
2-FCV-1-113	≤ 20 Seconds	4.53
2-FCV-1-114	≤ 20 Seconds	9.08

3.5.3 The closing modulation time for each steam dump control valve is less than 20 seconds upon the receipt of a 95% to 5% control signal step change.

Review criteria were met as delineated below:

Steam Dump Valve	Closed Stroke Timing Requirement	Actual Closed Stroke Timing
2-FCV-1-103	≤ 20 Seconds	5.32
2-FCV-1-104	≤ 20 Seconds	5.16
2-FCV-1-105	≤ 20 Seconds	9.61
2-FCV-1-106	≤ 20 Seconds	5.22
2-FCV-1-107	≤ 20 Seconds	7.18
2-FCV-1-108	≤ 20 Seconds	5.36
2-FCV-1-109	≤ 20 Seconds	8.48
2-FCV-1-110	≤ 20 Seconds	7.96
2-FCV-1-111	≤ 20 Seconds	5.38
2-FCV-1-112	≤ 20 Seconds	5.48
2-FCV-1-113	≤ 20 Seconds	6.91
2-FCV-1-114	≤ 20 Seconds	11.92

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

3.6 Section 6.2 (Static Valve Timing -Trip)

3.6.1 All of the steam dump control valves trip open in ≤ 3 seconds following a simulated HI-HI T_{avg} signal.

Review criteria were met as delineated below:

Steam Dump Valve	Closed Stroke Timing Requirement	Actual Closed Stroke Timing
2-FCV-1-103	≤ 3 Seconds	1.23
2-FCV-1-104	≤ 3 Seconds	1.57
2-FCV-1-105	≤ 3 Seconds	2.41
2-FCV-1-106	≤ 3 Seconds	1.92
2-FCV-1-107	≤ 3 Seconds	2.46
2-FCV-1-108	≤ 3 Seconds	2.11
2-FCV-1-109	≤ 3 Seconds	1.77
2-FCV-1-110	≤ 3 Seconds	2.09
2-FCV-1-111	≤ 3 Seconds	1.85
2-FCV-1-112	≤ 3 Seconds	1.46
2-FCV-1-113	≤ 3 Seconds	1.61
2-FCV-1-114	≤ 3 Seconds	1.85

3.6.2 All of the steam dump control valves trip closed in ≤ 5 seconds following a simulated block signal.

Review criteria were met as delineated below:

Steam Dump Valve	Closed Stroke Timing Requirement	Actual Closed Stroke Timing
2-FCV-1-103	≤ 5 Seconds	1.20
2-FCV-1-104	≤ 5 Seconds	1.42
2-FCV-1-105	≤ 5 Seconds	1.52
2-FCV-1-106	≤ 5 Seconds	1.70
2-FCV-1-107	≤ 5 Seconds	1.68
2-FCV-1-108	≤ 5 Seconds	1.41
2-FCV-1-109	≤ 5 Seconds	1.97
2-FCV-1-110	≤ 5 Seconds	1.89
2-FCV-1-111	≤ 5 Seconds	1.91
2-FCV-1-112	≤ 5 Seconds	1.45
2-FCV-1-113	≤ 5 Seconds	1.70
2-FCV-1-114	≤ 5 Seconds	1.74

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

3.7 Section 6.3 (Steam Pressure Controller)

3.7.1 After varying reactor power, the steam pressure controller controls steam header pressure at setpoint (± 25 psi) within nine minutes (three reset time constants).

The steam pressure controller controlled steam header pressure at setpoint (± 25 psi) within nine minutes after varying reactor power. Steam header pressure remained within the setpoint throughout the transient.

3.8 Section 6.4 (Plant Trip Controller)

3.8.1 Before and after varying reactor power, the plant trip controller demand signal remained within 2.0% of the calculated demand signal.

The plant trip controller demand signal remained within 2.0% of the calculated demand signal before and after varying reactor power.

Note: There were no Review Criteria for Sections 6.5 and 6.6.

4.0 Problems

[1] CR 1122945: During performance of WO 117441539, a jumper was placed on the wrong terminal point in Step 4.3.2[9] and an unexpected alarm was received in the Main Control Room. Work was stopped and the CR was initiated. Urgent Change, UC-1, was made to 2-PAT-5.1 Revision 2 to select a more accessible terminal point and labeled the terminal point for the jumper placement.

[2] CR 1123150: 2-FCV-1-105 failed to re-close following the trip open test. The CR was written to troubleshoot. The needle valve between the positioner and the diaphragm on 2-FCV-1-105 was found to be closed. After verifying that the needle valve should be in the open position, the needle valve was opened and 2-FCV-1-105 closed as expected. All other steam dump needle valves were verified to be in the open position.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

[3] CR 1124648: During performance of the steam dump sequence test in Step 6.1.2 [2B], the Data Sheet 2 steam dump valve position indications were not met at the three demand levels. The CR initiated WO 117506695 which was implemented, calibrating the steam dump controllers. In addition, the upper limit switch for 2-FCV-1-103 was found to be sticking and it was replaced under the work order. The Step 6.1.2 [2B], sequence test was then re-performed and the Data Sheet 2 steam dump valve positions were not met again at similar demand positions as the first performance.

The position indications that were not met in both sequence tests were associated with valve 20% open and 80% open positions. These positions are close to where the open and closed limit switches actuate to turn on or off the red and green position indication lights. It was determined that the steam dump valve sequence was acceptable because in each sequence test each bank of valves were full open before the next bank began to open and each valve modulated as required between full open and full closed demand positions. Data Sheet 2 was removed from the Review Criteria in Revision 3 to prevent additional unwarranted conditional reports and repairs.

[4] CR 1124788 and CR 1127374: During performance of the steam dump valve modulation stroke timing test in Section 6.1.3 through 6.1.6, the greater than 12 second valve stroke time requirement was not met.

It was determined that the greater than 12 second stroke time is not a requirement in any design documentation. In addition, there are no plant procedures that set up the valves to ensure a greater than 12 second modulation stroke time. This criteria was removed from 2-PAT-5.1 in Revision 3.

[5] CR 1174915: During transfer of steam dump control to the loss of load controller, a diverging oscillation was observed in the loss of load controller response. The loss of load controller response was found to be proper for the plant input signals to the controller and the high gain settings of the controller. The evaluation of the loss of load controller was documented in Westinghouse Letter LTR-SCS-16-23.

7.1.1 Dynamic Automatic Steam Dump Control (2-PAT-5.1) (continued)

[6] Urgent Change UC-1 to 2-PAT-5.1 Revision 0005 revised Step 6.5.2[4] and Acceptance Criteria 5.1.3[A] to verify the loss of load controller responds properly for the plant input signals to the controller. Urgent Change UC-1 revised the remainder of the loss of load transient testing and evaluation in Sections 6.5.2 and 6.5.3. Acceptance Criteria 5.1.3[B] and Review Criteria 5.2.5 were deleted. In addition, the Westinghouse test scoping document WAT/WBT-SU-2.8.5 Acceptance Criteria was revised to verify the loss of load controller responds properly. Also a SAR Change Package No. U2-021 was approved and issued that revised Chapter 14 Table 14.2-2 Sheet 28 to verify the load rejection controller responds properly. These changes were based on the Westinghouse Letter LTR-SCS-16-23 which documented the proper response of the load rejection controller.

[7] CR 1170159: 2-FCV-1-108 did not initially meet the trip open stroke time Acceptance Criteria of 3 seconds. A volume booster adjustment was made to 2-FCV-1-108 under WO 117826339 and the valve met the trip open stroke time when re-tested.

[8] CR 1170319: Piping vibration at 2-FCV-1-111 did not meet Acceptance Criteria during the stroke test. Civil Design Engineering evaluated the piping response and found it acceptable.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3)

This test was performed as part of test sequence 2-PAT-5.0, Test Sequence for 30% Plateau. The test began on 5/14/16 and completed on 6/15/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Demonstrate the proper operation and automatic response of the Steam Generator Level Control System for each Steam Generator during steady-state operation.

1.2 Satisfy, in part, the requirements of UFSAR Table 14.2-2, Sheet 30, Automatic Steam Generator Level Control Test Summary.

2.0 Test Methods

The UFSAR requires tests be performed at various power levels from 5% through 100% reactor power. This PAT tested low power aspects of the UFSAR requirement. For Section 6.1 the plant was in Mode 1 at less than 10% power with the main turbine not synchronized to the grid. For Section 6.2 the plant was in Mode 1 at approximately 30% power after the MFW Forward Flush/Back Flush Heatup had been completed. The test was performed in conjunction with maintenance work order activities to collect data needed to calibrate and tune feedwater control system components.

Actual testing in Section 6.1, Feedwater Bypass Control Valves, was started on dayshift 5/25/16 and completed on nightshift 5/26/16. All Acceptance and Review Criteria were met for Section 6.1, with no additional tuning of the Feedwater Bypass Control Valve Controllers being necessary.

Section 6.2, Transfer From Bypass To MFW Reg Valves, was initiated on 6/12/16. During the performance of Section 6.2.1, Transfer From Bypass To MFW Reg Valve For SG No. 1, the Steam Generator level was not stable within the required + 2% during the 10 minute monitoring period. The decision was made to continue testing in accordance with Section 6.2.4 for SG No. 4 and perform a re-test of Section 6.2.1 at a later time. During the performance of Section 6.2.4, Transfer From Bypass To MFW Reg Valve For SG No. 4, the Main Feedwater Reg Valve, 2-FCV-003-0103, did not respond to a 30% demand. It was determined that the air line to the MFW Reg Valve for SG No. 4 was leaking. WO 117904374 was written and performed to repair the leak.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

On 6/13/16, Re-Test #1 was performed for Section 6.2.1, Transfer From Bypass To MFW Reg Valve For SG No. 1. All Acceptance Criteria were met for SG No. 1 MFW Reg Valve with no adjustments being made. Repairs to the SG No. 4 air line were completed on 6/13/16 under WO 117904374, and testing was resumed for the SG No. 4 MFW Reg Valve. All four MFW Reg Valves successfully met the Acceptance Criteria upon completion of testing in Section 6.2. At the conclusion of testing, CR 1181278 was written to address areas of concern.

3.0 <u>Test Results</u>

All Acceptance/Review Criteria were met or resolved as delineated below.

<u>Acceptance Criteria</u>

3.1 Section 6.1 (Feedwater Bypass Control Valves)

3.1.1 The indicated steam generator level undershoot was less than 4.0% below the final setpoint following automatic recovery from high steam generator level.

The steam generator level undershoot ranged from 1% to 2% below the setpoint for all four Steam Generators, following automatic recovery from high steam generator level, which met the required Acceptance Criteria.

3.1.2 The indicated steam generator level overshoot was less than 4.0% above the setpoint following automatic recovery from low steam generator level.

The steam generator level overshoot ranged from -1% to 0% above the setpoint for all four Steam Generators, following automatic recovery from high steam generator level, which met the required Acceptance Criteria.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

3.2 Section 6.2 (Transfer from Bypass to MFW Control Valves)

3.2.1 Indicated steam generator level returned to and remained within ±2% of the program level within 10 minutes following the transfer of level control to the Main Feedwater Reg. Valves in automatic.

All four steam generator level indications returned to and remained within ±2% of the program level within 10 minutes following the transfer of level control to the Main Feedwater Reg. Valves in automatic. CR 1181278 was written due to questions regarding the wording of the Acceptance Criteria in 2-PAT-5.3 and is discussed under Problems.

3.2.2 Demand signal oscillations for each of the Main Feedwater Reg. Valves were less than ±6.0% during steady state operation.

All four Main Feedwater Reg Valves exhibited less than ±6% oscillation of the Main Feedwater Reg Valves in Auto:

- SG #1 - 2.17%
- SG #2 - 1.24%
- SG #3 - 2.81%
- SG #4 - 2.44%

CR 1181278 also addressed an issue with the procedure not being clear on the time data is recorded. This is discussed under Problems.

3.2.3 Feedwater flow oscillations to each Steam Generator were less than ±6.0% during steady state operation.

The feedwater flow oscillations to each Steam Generator are documented below:

- SG #1 - 3.41%
- SG #2 - 6.39%
- SG #3 - 5.59%
- SG #4 - 4.75%

The feedwater flow oscillations to SGs # 1, 3, and 4 met the Acceptance Criteria of less than ±6.0% during

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

steady state operation. CR 1181278 discussed in Problems addresses #2 SG data as acceptable.

Review Criteria

3.3 Section 6.1 (Feedwater Bypass Control Valves)

3.3.1 Indicated Steam Generator level returned to and remained within ±2% of the level setpoint within 37.5 minutes following automatic recovery from high Steam Generator level.

The indicated Steam Generator level returned to and remained within ±2% of the level setpoint within 37.5 minutes following automatic recovery from high Steam Generator level as shown below:

- SG #1 - 14 minutes
- SG #2 - 13 minutes
- SG #3 - 15 minutes
- SG #4 - 13 minutes

3.3.2 Indicated Steam Generator level returned to and remained within ±2% of the level setpoint within 37.5 minutes following automatic recovery from low Steam Generator level.

The indicated Steam Generator level returned to and remained within ±2% of the level setpoint within 37.5 minutes following automatic recovery from low Steam Generator level as shown below:

- SG #1 - 10 minutes
- SG #2 - 17 minutes
- SG #3 - 15 minutes
- SG #4 - 15 minutes

3.4 Section 6.2 (Transfer from Bypass to MFW Reg. Valves)

3.4.1 The Main Feedwater Reg. Valve position was between the minimum and maximum positions given in Figure 1 of 2-PAT-5.3 for the specific loop Main Steam Flow.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

Data indicated that the Main Feedwater Reg Valve positions exceeded the maximum positions given in Figure 1 of 2-PAT-5.3:

- Reg Valve 2-FCV-3-35 - 41.0%
- Reg Valve 2-FCV-3-48 - 45.4%
- Reg Valve 2-FCV-3-90 - 40.9%
- Reg Valve 2-FCV-3-103 - 40.9%

CR 1181278 was generated and Engineering was requested to evaluate the data and provide recommendations. Engineering's recommendation was to proceed with Power Ascension Testing to the 50% plateau.

3.4.2 Indicated Steam Generator level was within ±2% of the program Level within 10 minutes following Main Feed Reg. Valve being placed in AUTO and subsequent stable conditions steady state operations.

Data verified that the indicated Steam Generator levels were within ±2% of the program level within 10 minutes following Main Feed Reg. Valve being placed in AUTO and subsequent stable conditions for steady state operations.

3.4.3 The Main Feedwater Header Pressure oscillations were less than 108 psi (peak-to-peak) during steady state operations. (This limit was based on ±3.0% of the instrument span of 1800 psi).

The Main Feedwater Header Pressure oscillations were less than 108 psi (peak-to-peak) during steady state operations as shown below:

- Main Feedwater Header Pressure Oscillation For SG #1 - 5 psi
- Main Feedwater Header Pressure Oscillation For SG #2 - 8 psi
- Main Feedwater Header Pressure Oscillation For SG #3 - 8 psi
- Main Feedwater Header Pressure Oscillation For SG #4 - 4 psi

3.4.4 The Actual (measured) ΔP was within 25.0 psi of the Program ΔP during steady state operation.

The actual (measured) ΔP was 0.8 psi which met the Acceptance Criteria of being within 25.0 psi of the Program ΔP during steady state operation.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

4.0 Problems

[1] CR 1181278 was written due to questions regarding the wording of the Acceptance Criteria in 2-PAT-5.3. A comparison of 5.1.2 Acceptance Criteria for Section 6.2 (Transfer from Bypass to MFW Control Valves) to the Westinghouse document WBT-D-4709 (LTR-PCSA-14-31) confirmed that the Acceptance Criteria in 2-PAT-5.3 were written correctly. The steps in the body of the procedure to perform the test and verify the Acceptance Criteria were also reviewed with the originator of the CR. It was determined that the procedure for this section of the test was written correctly and neither the test nor the results were invalidated by the concerns in the CR.

CR 1181278 documented another concern which stated "The procedure is not clear if the performer looks at the data before or after a time. The procedure should say AFTER, because that is the approximate time that the main feedwater is transferred into Auto. With clarification, all Acceptance Criteria are met." A review of Data Sheet 11 revealed that for each Main Feedwater Reg Valve, the column to record data contains notation which states "Data from time in Step 6.2.X[18]". Step 6.2.X[18] recorded the end time for the 10 minute monitoring period. It would have been better if Data Sheet 11 would have stated "Data from monitoring period in Steps 6.2.X[16] through 6.2.X[18]". Additionally, during the review, it was discussed that the Acceptance Criteria was to monitor the Demand Signal oscillations for each of the Main Feedwater Reg Valves; however, the test kept the Reg valves in Manual instead of Auto during this portion of testing. Fortunately, the Test Coordinators collected the appropriate data with the valves in Manual, then swapped the controller position to Auto, as allowed by the procedure, and collected the appropriate data in this condition. The data was analyzed with the valves in AUTO and it was determined that the Acceptance Criteria were met. Since the data was also collected with the valves in AUTO and the Acceptance Criteria were met, there was no need to re-perform this section of the test.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

The data recorded in the test for the feedwater flow oscillations to SG #2 was above the acceptable limit of 6% for flow oscillations; however, CR 1181278 states "The feedwater was controlled and did not oscillate. However, the maximum deviation was about 6.39% of average flow." A review of the data indicated that the feedwater flow to SG #2 started off with a deviation of >6%; however, the controller brought the flow to within an acceptable range in a steady manner and maintained an acceptable flow rather than oscillating for a period of time (see Figure 7.1.2-1) The fact that the feedwater flow stabilized within a range which was less than 6% without oscillating meets the intent of the Acceptance Criteria.

CR 1181278 also requested Engineering to evaluate data on one Review Criteria and provide recommendations. Engineering's recommendation was to proceed with Power Ascension Testing to the 50% plateau for the following Review Criteria:

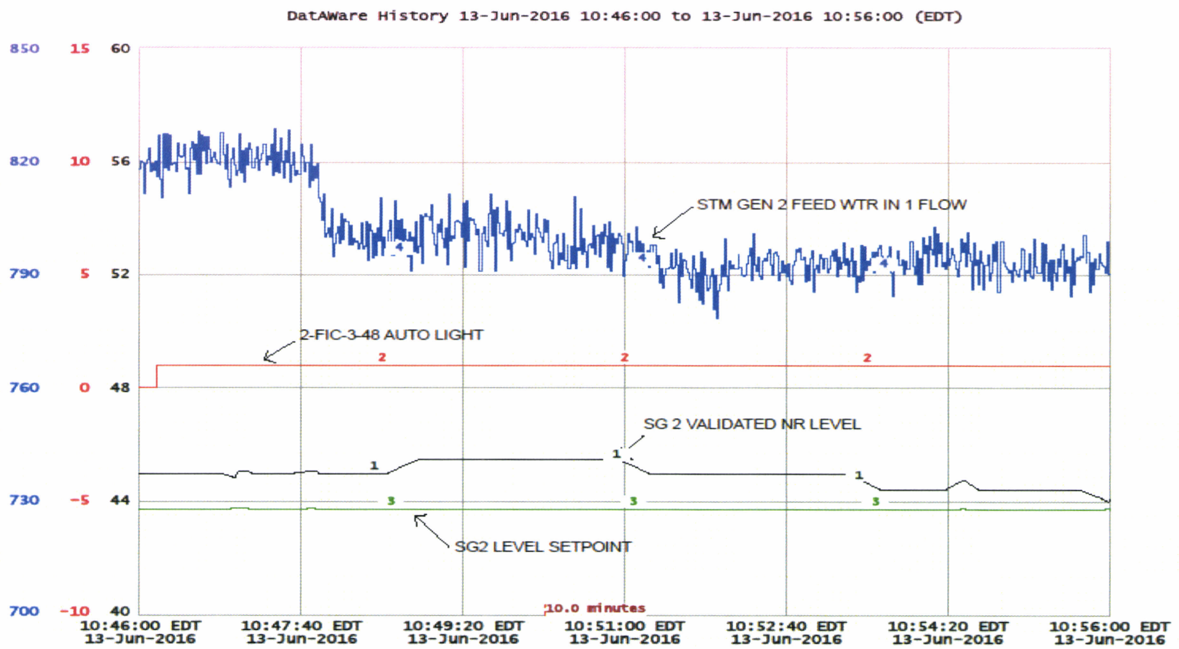
- Section 6.2, MFW Reg Valves were not between the minimum and maximum positions required

[2] WO 117904374 was initiated to repair a leak on the air line to #4 Steam Generator MFW Reg. Valve. The valve was retested after repairs and passed Acceptance Criteria.

7.1.2 Automatic Steam Generator Level Control Transients at Low Power (2-PAT-5.3) (continued)

FIGURE 7.1.2-1

SG #2 FW Flow Oscillation



Point ID/Description	Low-Y	Hi-Y	Units
(1) DCS0046 (W2:U2) SG 2 VALIDATED NR LEVEL	40	60	%
(2) FIC0030048J (W2:U2) 2-FIC-3-48 AUTO LIGHT	-10	15	
(3) DCS0074 (W2:U2) SG 2 LEVEL SETPOINT	40	60	%
(4) F0423A (W2:U2) STM GEN 2 FEED WTR IN 1 FLOW	700	850	KBH

7.1.3 Calibration of Steam and Feedwater Flow Instruments at 30% Power (2-PAT-5.4)

This test was performed with the plant stable at approximately 30% Power as part of 2-PAT-5.0, Test Sequence for 30% Plateau. The test began on 6/13/16 and was complete on 6/15/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Verify the output of the eight feedwater flow transmitters for "zero" output with minimal feedwater flow, collect data for determining the new calibration spans for the steam flow transmitters

1.2 Verify the calibration of the feedwater and steam flow transmitters, by comparing indicated flows between the Main Control Board Indicators, the Protection System, and the Control System.

1.3 Satisfy, in part, the 30% objective in the UFSAR Table 14.2-2, Sheet 21, Calibration Of Steam And Feedwater Flow Instrumentation At Power Test Summary.

2.0 Test Methods

At approximately 30% power, each feedwater flow transmitter was placed in bypass and verified for “zero” output.

At approximately 30% power, steam generator blowdown and tempering flow were isolated while data was collected. Steam generator blowdown and tempering flow were then reestablished and calculations/comparisons were performed.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

None

7.1.3 Calibration of Steam and Feedwater Flow Instruments at 30% Power (2-PAT-5.4) (continued)

Review Criteria

3.1 At zero DP, the output from each Feedwater Flow Transmitter and its associated loop reflect zero flow as demonstrated by the following criteria:

A. Computer Point: 0.000 KBH (-25.5 to 25.5 KBH)
B. Flow Test point: 0.200 Vdc (0.1858 to 0.2142 Vdc)
C. Computer Test Point: 0.200 Vdc (0.1858 to 0.2142 Vdc)
D. DP Test point: 0.1983 Vdc (0.1848 to 0.2118 Vdc)

The data below was collected and the output flow was verified within the Review Criteria requirements.

DESCRIPTION	STEAM GENERATOR 1		STEAM GENERATOR 2	
	2-FT-3-35A	2-FT-3-35B	2-FT-3-48A	2-FT-3-48B
Computer Point Flow (KBH) 5.2.A.1	5	1	1	1
Flow Test Point 5.2.A.2	0.20148	0.20028	0.20154	0.20055
Comp Test Point 5.2.A.3	0.20089	0.20032	0.20049	0.20029
DP Test Point Voltage 5.2.A.4	0.20269	0.19589	0.19763	0.19912

DESCRIPTION	STEAM GENERATOR 4		STEAM GENERATOR 3	
	2-FT-3-90A	2-FT-3-90B	2-FT-3-103A	2-FT-3-103B
Computer Point Flow (KBH) 5.2.A.1	4	2	-3	1
Flow Test Point 5.2.A.2	0.20104	0.20074	0.20056	0.20027
Comp Test Point 5.2.A.3	0.20067	0.20038	0.19956	0.20043
DP Test Point Voltage 5.2.A.4	0.19945	0.19749	0.19589	0.19623

3.2 The difference between the Feedwater Flow as measured in the Protection System and the Main Control Board Indicators is within ±5.0% of the rated flow.

Measured differences (% ERRORs) between -1.89% and +1.18%

3.3 The difference between the feedwater flow as measured in the Protection System and the Indicated Computer Feedwater Flow is within ±2.0% of rated flow.

Measured differences (% ERRORs) between -0.24% and +0.06%

7.1.3 Calibration of Steam and Feedwater Flow Instruments at 30% Power (2-PAT-5.4) (continued)

3.4 The difference between the feedwater flow as measured in the Protection System and the Feedwater Flow Signal used for flow control is within ±2.0% of rated flow.

Measured differences (% ERRORs) between -0.29% and -0.07%

3.5 The difference between the steam flow as measured in the Protection System and the Main Control Board Indicators is within ±5.0% of the rated flow.

Measured differences (% ERRORs) between -0.87% and +0.94%

3.6 The difference between the steam flow as measured in the Protection System and the Indicated Computer Steam Flow is within ±2.0% of rated flow.

Measured differences (% ERRORs) between -0.10% and 0.00%

3.7 The difference between the steam flow as measured in the Protection System and the Steam Flow Signal used for flow control is within ±2.0% of rated flow.

Measured differences (% ERRORs) between -0.23% and 0.00%

3.8 The difference between the feedwater flow as measured in the Protection System and the Steam Flow as measured in the Protection System is within ±5.0% of rated flow.

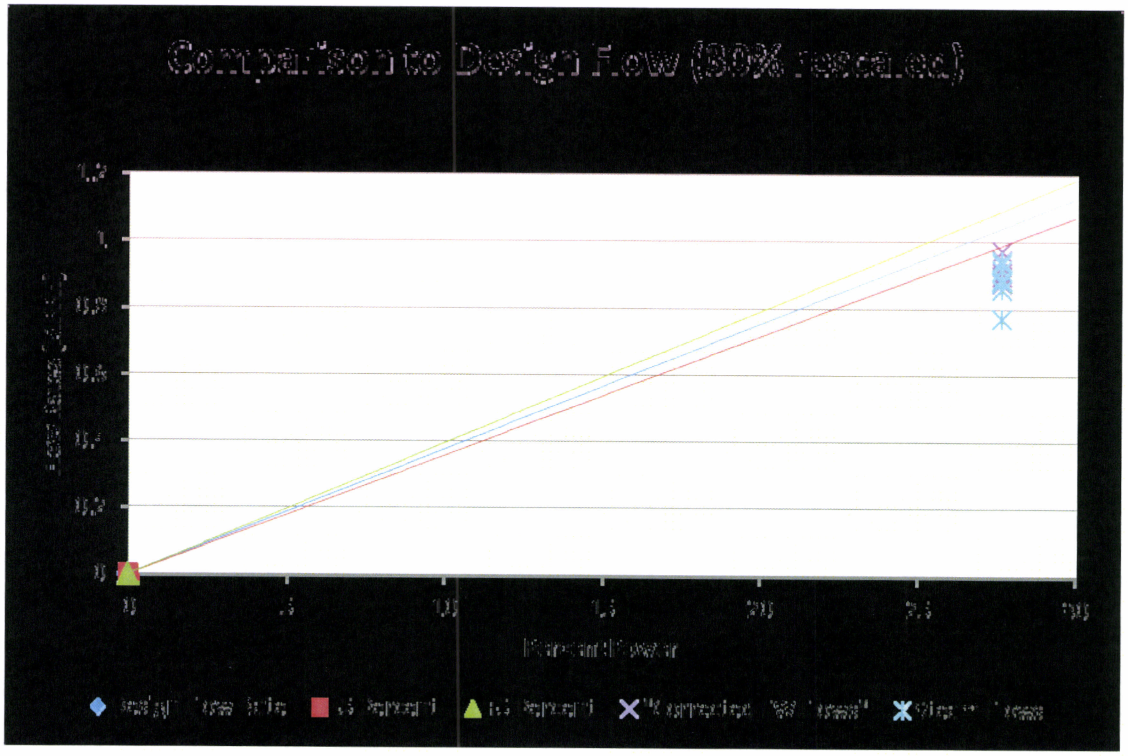
Measured differences (% ERRORs) between -2.62% and +4.45%

3.9 The difference between the feedwater flow as measured in the Control System and the Steam Flow as measured in the Control System is within ±5.0% of rated flow.

Measured differences (% ERRORs) between -2.57% and +4.56%

7.1.3 Calibration of Steam and Feedwater Flow Instruments at 30% Power (2-PAT-5.4) (continued)

Additionally, a comparison of the corrected feedwater flows and steam flows versus predicted design flow is provided:



4.0 <u>Problems</u>

There were no significant problems encountered during the performance of this test.

7.2 Test Sequence for 50% Plateau (2-PAT-6.0)

This test started on 5/30/16 and was completed on 7/16/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 In conjunction with 2-GO-4, Normal Power Operation, define the plant operational requirements and ensure those requirements were met in order to permit power escalation from 30% Rated Thermal Power (RTP) to 50%.

1.2 Specify the order of test performance at the 50% plateau.

The following PATs/PETs were sequenced for performance by 2-PAT-6.0:

- 2-PAT-1.4 * Pipe Vibration Monitoring
- 2-PAT-1.5 * Loose Parts Monitoring System
- 2-PAT-1.6 * Startup Adjustments of Reactor Control System
- 2-PAT-1.7 * Operational Alignment of Process Temperature Instrumentation
- 2-PAT-1.8 * Thermal Expansion of Piping Systems
- 2-PAT-1.10* Integrated Computer System (ICS)
- 2-PAT-1.11* RVLIS Performance Test
- 2-PAT-1.12* Common Q Past Accident Monitoring System
- 2-PAT-3.3 * RCS Flow Measurement
- 2-PAT-5.2 Turbine Generator Trip With Coincident Loss of Offsite Power Test
- 2-PAT-6.1 Automatic Reactor Control System
- 2-PAT-6.2 Automatic Steam Generator Level Control Transients at 50% Power
- 2-PAT-6.3 Calibration of Steam and Feedwater Flow Instruments at 50 % Power
- PET-301 * Core Power Distribution Factors
- PET-304 * Operational Alignment of NIS
- RCI-159 * Radiation Baseline Surveys

Note: * Indicates that the test is performed at multiple test plateaus. The description of the testing is documented in the section (plateau) in which it was completed.

7.2 Test Sequence for 50% Plateau (2-PAT-6.0) (continued)

2.0 Test Methods

Prerequisite actions for this Power Ascension Test (PAT) started on 5/30/16 and completed on 6/17/16 and included verification of the following major items:

- 2-PAT-5.0, Test Sequence for 30% Plateau, complete
- NPG-SPP-10.4, Reactivity Management Program, Reactivity Control Plans were developed to support the planned testing for this sequence
- Reactor power between 27% and 29% RTP with T_{avg}-T_{ref} mismatch ±1.5 °F or less
- RCS pressure is between 2220 to 2250 psig
- Section 6.5 of 2-PET-304, Operational Alignment of NIS, to adjust the Power Range High Flux Level Trip setpoints for testing at the 50% Plateau complete

Power increase to the 50% testing plateau was initiated on 6/17/16 at 11:40 and PAT testing in Section 6.1 of 2-PAT-6.0 was begun. On 6/20/17 at 15:37, U-2 Turbine tripped due to the loss of 2B Main Feedwater Pump and subsequently an automatic Reactor Trip occurred at 15:40 due to S/G levels reaching their low-low trip setpoint. The plant was stabilized in Mode 3.

Unit 2 re-entered Mode 2 on 6/23/16 at 17:37 and the reactor critical at 17:53. Mode 1 entry was made at 03:00 on 6/24/16. The U-2 generator was synchronized to the grid at 13:58.

A manual turbine trip was initiated on 6/26/16 at 09:45 due to a steam leak. Reactor power was reduced and the Unit entered Mode 2 at 11:44. At 15:26 the reactor was tripped manually and the unit stabilized in Mode 3.

Mode 2 was again entered on 7/2/16 at 03:00 with reactor criticality at 03:20. U-2 entered Mode 1 at 07:57 and was synchronized to the grid in the afternoon at 13:36. On 7/7/16 the 50% Plateau power level testing was reached and the 50% tests were commenced. Steady state testing included:

- 2-PAT-1.4, Pipe Vibration Monitoring, completed on 7/8/16 with all criteria met.
- 2-PAT-1.5, Loose Parts Monitoring System, was completed on 7/7/16 with all criteria met. CR 1171424 documents three channels removed from service.
- 2-PAT-1.6, Startup Adjustments of Reactor Control System, was completed on 7/8/16. This was data taking only with no Review or Acceptance Criteria at this plateau.

7.2 Test Sequence for 50% Plateau (2-PAT-6.0) (continued)

- 2-PAT-1.7, Operational Alignment of Process Temperature Instrumentation, was completed on 7/8/16 with all criteria met.
- 2-PAT-1.8, Thermal Expansion of Piping Systems, was field work complete on 7/8/16 with 2 issues referred to engineering for evaluation with Problem Report #4. Engineering review indicated it was acceptable to continue Power Ascension Testing.
- 2-PAT-1.10, Integrated Computer System (ICS), was completed on 7/8/16 with all criteria met.
- 2-PAT-1.11, RVLIS Performance Test, applicable sections were completed on 7/7/16 with all criteria met.
- 2-PAT-1.12, Common Q Post Accident Monitoring System, applicable sections were completed on 7/7/16 with all criteria met.
- 2-PAT-6.3, Calibration of Steam and Feedwater Flow Instruments at 50% Power, was completed on 7/8/16 with all criteria met.
- 2-PAT-3.3, RCS Flow Measurement, was completed for the 50% Plateau on 7/9/16 with all criteria met.
- RCI-159, Radiation Baseline Surveys, was completed for the 50% plateau on 7/10/16. No Acceptance or Review Criteria were associated with this procedure.

Transient tests were begun on 7/11/16 and included the following:

- 2-PAT-6.1, Automatic Reactor Control System, was completed on 7/13/16 with all criteria met.
- 2-PAT-6.2, Automatic Steam Generator Level Control Transients, was completed on 7/16/16 with all criteria met.
- 2-PAT-5.2, Turbine Generator Trip With Coincident Loss of Offsite Power Test, was completed on 7/14/16 with all criteria met except one Review Criteria. CR 1192287 was written to document Tcold going below the 547°F criteria.
- 2-PAT-1.4, Pipe Vibration Monitoring, for transient testing was completed on 7/14/16 with all criteria met.

2-PAT-1.2, Load Swing Test, originally scheduled for the 50% plateau, was revised to allow performance during 2-PAT-7.0 due to the inability of the turbine to be operated in IMP IN. Repairs to the circuitry were evaluated during the outage and a procedure revision was made to allow performance of the Load Swing Test in IMP OUT on the turbine controls.

7.2 Test Sequence for 50% Plateau (2-PAT-6.0) (continued)

Additionally, Engineering completed the following procedures or applicable sections during the steady state period, with no issues, to support their testing at the 50% Plateau:

- 2-TI-41 - Incore Flux Mapping
- 2-TRI-0-22 - PDMS Operability
- 2-SI-0-21 - Excore QPTR & Axial Flux Difference
- 2-PET-301 - Core Power Distribution Factors
- 2-SI-92-3 - Incore-Excore Cross Calibration Data
- 2-TI-7.020 - PDMS Calibration
- 2-PET-304 - Operational Alignment of NIS
- 2-TI-6 - Calorimetric Calibration
- 2-SI-0-20 - Hot Channel factors Determination
- 2-SI-92-2 - NIS Monthly Recalibration data
- 2-SI-0-22 - Incore QPTR

Details of the performance of each PAT procedure is contained in the individual summaries of the associated procedures as they are fully completed.

3.0 Test Results

All Acceptance/Review Criteria were contained within the tests sequenced by this test.

4.0 Problems

Problems encountered are addressed in the following discussions of each test sequenced by 2-PAT-6.0.

7.2.1 Turbine Generator Trip with Coincident Loss of Offsite Power Test (2-PAT-5.2)

This test was performed as part of 2-PAT-6.0, Test Sequence for 50% Plateau, and initiated in Mode 1. The test began pre-requisites on 7/7/16 and was field work completed on 7/14/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Demonstrate Unit 2 response to a turbine generator trip with a coincident loss of offsite power (LOOP) is in accordance with design.

1.2 Demonstrate that all four emergency diesel generators (EDG) automatically start, the Unit 2 EDGs connect to their respective shutdown board and provide power to the controls, indications, and equipment necessary to maintain Unit 2 in Hot Standby (Mode3) conditions for a minimum of 30 minutes.

1.3 Demonstrate that operators can control plant parameters using equipment available during a loss of offsite power.

1.4 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 33, Turbine Generator Trip With Coincident Loss Of Offsite Power Test Summary.

1.5 Provide the steps necessary to protect Unit 1 operations.

2.0 Test Methods

Initial conditions for Unit 2 include reactor power at approximately 30% of rated thermal power, the main generator synchronized to the TVA grid, and electrical load greater than or equal to 120 MWe. All four diesel generators were in their normal standby condition.

Unit 1 was in Mode 1 with alignment of the Unit 1 Reactor Coolant Pump Boards, Unit 1 Unit Boards, Common Boards and Shutdown Boards 1A-A and 1B-B energized from normal power sources, the USST's associated with Mode 1 operation.

The C-S CCS pump was aligned and in service to supply header 1B/2B in accordance with 0-SOI-70.01, "Component Cooling Water System". The automatic transfer of the 2A and 2C RCP boards to the A RCP Start bus was blocked and the automatic transfer of the 2B and 2D RCP boards to the B RCP Start bus was blocked. The automatic transfer of the 2A-A Shutdown board to the 2A-A Diesel Generator and the automatic transfer of the 2B-B Shutdown board to the 2B-B Diesel Generator were not blocked. The automatic transfer of the 2A-A Shutdown board to the D CSST and the automatic transfer of the 2B-B Shutdown board to the C CSST was blocked. The maintenance supplies to the Unit 2 Shutdown

7.2.1 Turbine Generator Trip with Coincident Loss of Offsite Power Test (2-PAT-5.2) (continued)

Boards were verified in the "racked down/removed" position. The 6.9 KV B common board was in its normal alignment. The B Common Board was not de-energized during the test to protect auxiliaries on both Units.

The Unit 2 Main Turbine was manually tripped. Following the turbine trip, Operations concurrently and immediately performed the following:

- Opened the normal power supply breaker to the 2A-A Shutdown Board. The board did not transfer to its alternate power source, resulting in a dead board condition. All four emergency diesel generators (EDGs) started as expected. The 2A-A EDG connected to and energized the 2A-A Shutdown Board.
- Opened the normal power supply breaker to the 2B-B Shutdown Board. The board did not transfer to its alternate power source, resulting in a dead board condition. The 2B-B EDG connected to and energized the 2B-B Shutdown Board as expected.
- Operations ensured the U2 Main turbine and U2 Main Generator tripped.

Following the Unit 2 generator trip, all four Unit 2 RCP Boards did not transfer to their alternate power source and remained de-energized. An automatic reactor trip of Unit 2 occurred when voltage was lost to the Unit 2 Reactor Coolant Pumps (RCPs). Operations then entered 2-E-0, Reactor Trip or Safety Injection.

With all four Unit 2 reactor coolant pumps de-energized, the RCS developed natural circulation conditions. Natural circulation parameters took longer than Unit 1 to establish due to the very low decay heat generated in the new core. The Unit 2 Main Steam Isolation Valves (MSIV's) were manually closed to support testing with a simulated Loss of Offsite Power configuration. Unit 2's Main Steam line pressure, steam generator pressure, and RCS temperature were maintained by the SG PORVs discharge to atmosphere. Auxiliary feedwater automatically started, and steam generator level trended to post trip setpoint conditions.

The test ran at least 30 minutes after the 2A-A and 2B-B 6.9kV Shutdown Boards were energized from their respective emergency diesel generators without restoring offsite power. Unit 2 was restored to a planned outage upon test completion at the direction of the Shift Manager.

7.2.1 Turbine Generator Trip with Coincident Loss of Offsite Power Test (2-PAT-5.2) (continued)

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

Acceptance Criteria

3.1 The 2A-A Diesel Generator automatically starts and connects to 2A-A Shutdown Board following the Loss of Offsite Power transient.

2A-A Diesel Generator automatically started and connected to its respective shutdown board.

3.2 The 2B-B Diesel Generator automatically starts and connects to 2B-B Shutdown Board following the Loss of Offsite Power transient.

2B-B Diesel Generator automatically started and connected to its respective shutdown board.

3.3 The Unit 2 Pressurizer Safety Valves do not open during the test.

Pressurizer Safeties did not open during the test.

3.4 The Unit 2 Steam Generator Safety Valves do not open during the test.

Steam Generator Safety Valves did not open during the test.

3.5 A Unit 2 Safety Injection is not initiated during the test.

No safety injection was initiated during testing.

3.6 Hot standby (Mode 3) conditions on Unit 2 are maintained for at least 30 minutes after the 2A-A and 2B-B 6.9kV Shutdown Boards are energized from respective emergency diesel generators without restoring offsite power.

Hot Standby (Mode 3) conditions were maintained for at least 30 minutes after 2A-A and 2B-B were energized for their respective emergency diesel generators without restoring offsite power.

7.2.1 Turbine Generator Trip with Coincident Loss of Offsite Power Test (2-PAT-5.2) (continued)

3.7 The 1A-A Diesel Generator automatically starts but does not connect to the 1A-A Shutdown Board following the Loss of Offsite Power transient.

1A-A Diesel Generator automatically started but did not connect to the 1A-A Shutdown Board.

3.8 The 1B-B Diesel Generator automatically starts but does not connect to the 1BB Shutdown Board following the Loss of Offsite Power transient.

1B-B Diesel Generator automatically started but did not connect to the 1B-B Shutdown Board.

Review Criteria

3.9 The following Unit 2 parameters were maintained within their respective limits for at least 30 minutes immediately after de-energizing 2A-A and 2B-B Shutdown Boards, using equipment available with offsite power removed from Unit 2:

3.9.1 RCS Cold Leg Temperature (547°F to 560°F and changing at a rate less than 50°F in one hour)

This criteria was not met. RCS Cold Leg Temperatures reduced below the minimum temperature of 547 degrees during the 30 minute period. The rate of change was less than 50 degrees in one hour. CR 1192287 documented this issue and was due to Turbine Driven AFW cooling since reactor decay heat was minimal and no RCPs in service.

3.9.2 Pressurizer Level (17% to 50%)

Pressurizer level maintained between 25% and 34% during the test period.

3.9.3 Pressurizer Pressure (2000 psig to 2335 psig)

Pressurizer Pressure maintained between 2119 and 2244.8 psig during the test period.

3.9.4 Steam Generator Levels (17% to 60% narrow range and either constant or trending toward 38% of narrow range)

Steam Generator narrow range level maintained between 30% and 39% during the test period.

7.2.1 Turbine Generator Trip with Coincident Loss of Offsite Power Test (2-PAT-5.2) (continued)

4.0 Problems

[1] CR 1192287 documented the Review Criteria was not met when the RCS cold leg temperature decreased below the minimum criteria of 547°F. This deficiency is attributed to the Turbine Drive Auxiliary Feedwater Pump steam supply source cooling the loop as it supplied AFW to the steam generators. With minimal reactor decay heat and no RCPs running, the loop temperature was not maintained above the minimum criteria.

[2] CR 1192023 was written to address the observation that the 2A-A Diesel Generator appeared to be slower than expected in tying on to its shutdown board. This was neither a Review or Acceptance Criteria for this test. Subsequent review by plant staff did indicate the 2A-A Diesel Generator did not tie onto the board within the Technical Specification limit and was declared inoperable. The diesel generator was repaired by plant maintenance and had no impact on meeting the PAT criteria as delineated in UFSAR Chapter 14, Table 14.2.2, Sheet 33.

7.2.2 Automatic Reactor Control System (2-PAT-6.1)

This test was performed at the 50% test plateau as directed by 2-PAT-6.0, Test Sequence for 50% Plateau. Testing was started on 6/18/16 and field work completed on 7/12/16.

1.0 Test Objectives

The objectives of this test were to:

1.1 Demonstrate the ability of the Automatic Rod Control System to maintain the average RCS temperature (T_{avg}) within acceptable limits during both steady-state and transient conditions.

1.2 Satisfy the requirements of UFSAR Table 14.2-2, Sheet 31, Automatic Reactor Control System Test Summary.

2.0 Test Methods

With the Reactor Control System (i.e. Rod Control) in manual and the Reactor Coolant System (RCS) at steady state conditions, Rod Control was placed in automatic to demonstrate that steady state conditions could be maintained.

Subsequently, with Rod Control in manual, T_{avg} was varied from the Reference Temperature (T_{ref}) by approximately +6 °F(+5°F to +7°F), by manually changing the position of Control Bank D with no deliberate turbine load change. Rod Control was then placed in automatic to demonstrate the ability to restore and stabilize T_{avg} to within a ±1.5°F dead band from T_{ref} via proper positioning of Control Bank D. The same test was also performed for a T_{avg} change of approximately -6°F (-5°F to -7°F) relative to T_{ref}.

The test was performed with reactor power approximately 45% to 47% of Rated Thermal Power (RTP) and RCS average temperature, pressurizer level and steam generator levels on program. The initial T_{avg} - T_{ref} mismatch was within ±1°F and RCS pressure was between 2200 to 2250 psig.

3.0 Test Results

All Acceptance/Review Criteria were met or resolved as delineated below.

No control system settings were changed based on the performance of this test.

Figures 7.2.2-1 through 7.2.2-9 depict the performance results of the automatic control systems.

7.2.2 **Automatic Reactor Control System (2-PAT-6.1)** (continued)

Acceptance Criteria

3.1 No manual operator action or intervention is required to return the plant to stable conditions (i.e., auctioneered RCS T_{avg} within $\pm$1.5°F of T_{ref}) for both steady-state and transient conditions.

No manual operator action or intervention was required.

3.2 For steady-state operation, and for both increasing and decreasing T_{avg} temperature transients, the Automatic Rod Control System responds properly to automatically position control rods and return auctioneered RCS T_{avg} to within $\pm$1.5°F of T_{ref} when the ARCS is placed in AUTO control mode.

Rod Control properly responded to steady state and transient conditions to return T_{avg} to within $\pm$1.5°F of T_{ref}.

Review Criteria

3.3 Pressurizer pressure tracks the response of auctioneered T_{avg} during the T_{avg} transient tests and is controlled back to approximately 2235 psig due to automatic pressurizer pressure control.

Pressurizer pressure tracked the response to Tavg and controlled back to approximately 2235 psig.

3.4 Pressurizer level and level setpoint track the response of auctioneered T_{avg} during the T_{avg} transient tests due to automatic pressurizer level control.

Pressurizer level and level setpoint tracked the response to T_{avg}.

7.2.2 **Automatic Reactor Control System (2-PAT-6.1)** (continued)

4.0 Problems

[1] CR 1190719 was written for two procedure deficiencies on 2-PAT-6.1, Automatic Reactor Control System.

Steps 6.2[8] and 6.3[12] said to ENSURE the passive summer indicated 72 steps/min. The passive summer does not indicate rod speed. The passive summer indicates an error signal in Degrees F. Steps 6.2[8] and 6.3[12] should have ENSURED the passive summer indicated +5ºF and - 5ºF, respectively. The error was identified, discussed by PAT and Operations, CTL entry entered, and the test was continued. The +/- 5 degrees was verified during performance. Also Steps 6.2[10] and 6.3[14] verified that the rod speed was 72 steps/min at the time the rods were placed to auto. "Step 6.3[3]l." was a typo and should have been deleted.