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SUBJECT: Submits cycle 12 startup rept for plant per TS.Encl repts provide descriptions of startup & power ascension testing activities.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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December 4, 1996
GO2-96-235

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NO. NPF-21**
CYCLE 12 STARTUP REPORT

References:
1. Letter GO2-95-228, dated October 26, 1995, JV Parrish (SS) to NRC, "Request for Amendment to Technical Specifications, Reactor Recirculation System Adjustable Speed Drive Upgrade"
2. Letter GI2-96-183, dated July 17, 1996, TP Gwynn (NRC) to JV Parrish (SS), "NRC Inspection Report 50-397/96-07"

The Supply System hereby submits the Cycle 12 Startup Report for WNP-2 in compliance with Technical Specifications. WNP-2 Technical Specification 6.9.1.1 requires submittal of a summary report following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Spring 1996 (R-11) maintenance and refueling outage commenced on April 13, 1996 and was completed on June 22, 1996. However, power ascension was protracted because of startup and power ascension testing. As a result, full (100%) reactor power for Cycle 12 commercial operation was not achieved until September 5, 1996.

During the R-11 outage, the annual refueling activities included installation of 104 SVEA-96 fuel assemblies designed and manufactured by ASEA Brown Boveri/Combustion Engineering (ABB-CE). These 104 fuel assemblies are of a different design than previously installed at WNP-2 and ABB-CE is a different fuel supplier. WNP-2 is a General Electric (GE) "BWR-5" design utilizing a total of 764 fuel assemblies. The reactor core is currently composed of a mixture of active fuel assemblies manufactured by Siemens Power Corporation (SPC) and ABB-CE. In

addition to installation of the new fuel, the R-11 outage scope also included installation of new Digital Feedwater (DFW) and Reactor Recirculation (RRC) pump adjustable speed drive (ASD) control systems. Installation of the DFW and ASD modifications could potentially alter the nuclear, thermal, and hydraulic performance of the plant.

The attached reports provide descriptions of the startup and power ascension testing activities following installation of the new ABB-CE fuel and the DFW and ASD modifications.

Should you have any questions or desire additional information regarding this matter, please call me or Ms. L. C. Fernandez at (509) 377-4147.

Respectfully,

PM Swank For PRB per telecon

P. R. Bemis
Vice President, Nuclear Operations
Mail Drop PE23

Attachment
1. ABB-CE Fuel Startup and Power Ascension Report
2. DFW/ASD Startup and Power Ascension Report

cc: LJ Callan - NRC RIV
KE Perkins, Jr. - NRC RIV, Walnut Creek Field Office
NS Reynolds - Winston & Strawn
TG Colburn - NRR
DL Williams - BPA/399
NRC Sr. Resident Inspector - 927N

ABB-CE REACTOR FUEL

As discussed in the cover letter, the R-11 outage refueling activities included installation of the 104 SVEA-96 fuel assemblies designed and manufactured by ABB-CE. The testing associated with the R-11 refueling process were performed at various stages during the outage, as well as before and after the outage. Testing included POWERPLEX Installation Acceptance Testing, Shutdown Margin Determination Testing, Reactivity Anomaly Evaluation Testing, Control Rod Functional Testing, Core Power Symmetry Testing, and Local Power Range Monitor (LPRM) and Traversing In-Core Probe (TIP) Response to Control Rod Motion Testing.

Refueling Activities

Refueling of the reactor for the R-11 outage was accomplished by a full core off-load and re-load. During fuel assembly insertion, the Source Range Monitor (SRM) count rate was monitored and tracked using inverse count rate ratio plots to ensure that criticality was not being approached. The refueling activities progressed normally with no unexpected events or interruptions.

Upon completion of core alterations, a full core verification was performed to visually verify fuel bundle identification numbers, locations, and orientations. The full core verification was simultaneously and independently reviewed by a second team. The refueling bridge mast was used as a movable gage to confirm that fuel assemblies were properly seated in the fuel support pieces. No deviations were identified by this process and it was concluded that all fuel assemblies were properly seated.

POWERPLEX Installation Acceptance Testing

A new version of the on-line core monitoring system, POWERPLEX II CMSS, was installed in the plant computer to support core surveillance. This new version of the POWERPLEX system uses SPC's MICROBURN methodology which incorporated the ABB-CE minimum critical power ratio correlation required to monitor ABB-CE fuel. As part of the installation, the POWERPLEX Cycle 12 input deck was reviewed for accuracy. The review included all cycle loading information, fuel design data, and limits as specified in the Cycle 12 Core Operating Limits Report (COLR).

The installation of POWERPLEX II CMSS and the successful performance of the installation acceptance testing were completed in June 1996, prior to initial core criticality. Based on the testing described below, it was concluded that this new version of the POWERPLEX system as installed met the requirement to monitor SPC and ABB fuel limits.

POWERPLEX II CMSS had been tested previously on a separate, off-line computer during Cycle 11. These Cycle 11 tests were performed to verify that plant data required by the POWERPLEX system are properly collected, transmitted, and processed. Some minor software problems that could have affected output accuracy were identified and resolved during this phase of testing.

The installation acceptance testing performed in June 1996 was divided into 32 sections (chapters), each addressing specific POWERPLEX functions. Each chapter may contain one or more individual calculations. Parameters compared for each MICROBURN calculation are: Critical Eigenvalue, Top Ten Core Thermal Limits (MAPRAT, MFLHGR, MFLCPR), and reactor power distribution (core average axial profile and fuel assembly average and maximum for each fuel type). The results show excellent agreement for each chapter.

Three chapters are dedicated to verifying the MICROBURN-B code.

Chapter 3 - MICROBURN-B Installation: This chapter verifies the MICROBURN program installation. The chapter contains 201 MICROBURN calculations based on different plant and core operating configurations. For each calculation, the results were identical to those provided by SPC.

Chapter 5 - PREDICT/MON Verification: This chapter verifies the PREDICT and MON calculations. The chapter contains 28 MICROBURN calculations and, for each calculation, the results were compared to SPC values and were within the acceptance criteria.

Chapter 6 - THERMAL LIMITS Test: This chapter verifies the THERMAL LIMITS calculation options of the MICROBURN program. The chapter contains 69 MICROBURN calculations based on various combinations of fuel loading and thermal limit sets and types. For each calculation, the results were compared to SPC values and were within the acceptance criteria.

The remaining 29 chapters tested other POWERPLEX II CMSS functions. The results of these tests were satisfactory and within the acceptance criteria specified by SPC.

The Supply System performed several tests in addition to the vendor supplied tests to verify the POWERPLEX features that were added to support monitoring ABB-CE fuel. These tests included verifications of the ABB-CE Critical Power Ratio (CPR) correlation (XLS-96) and ABB-CE fuel preconditioning criteria. The results from the ABB-CE CPR correlation were compared to those supplied by ABB-CE for several reference cases developed using the ABB-CE CONDOR computer code. Table 1.1 below provides a comparison of the calculated CPRs. The differences are within the acceptable range.

Table 1.1: Calculated CPR Comparison

Core Conditions	CPR Calculated by ABB Methodology	CPR Calculated by SPC MICROBURN Methodology	Differences
100% power and 100% core flow	1.464	1.464	0
105% power and 106% core flow	1.446	1.446	0
80% power and 100% core flow	1.855	1.858	0.003
65% power and 50% core flow	1.598	1.600	0.002

Shutdown Margin Demonstration

An in-sequence shutdown margin (SDM) demonstration was performed during the first startup following the R-11 outage refueling activities. WNP-2 Technical Specification 3/4.1.1 requires that this SDM be equal to or greater than 0.38% ΔK/K, if the highest worth rod is analytically determined. The reactor went critical at Step 24 of Group 2 (Rod 18-47 at Notch 08) with a moderator temperature of 154.5 °F and an 84 second period. The SDM was demonstrated to be greater than 1.714% ΔK/K. An examination of the calculations in the Cycle 12 Startup and Operations Report showed that an additional SDM demonstration at some greater exposure would not be required.

Core Reactivity Anomaly Evaluation

WNP-2 Technical Specification 3/4.1.2 requires that the difference between the monitored and predicted core reactivity coefficient, K_{eff}, must not exceed 1.0% ΔK/K during the first startup following core alterations. This reactivity anomaly testing was performed during the SDM demonstration, comparing a calculated predicted cold reactivity coefficient with the measured value. The predicted K_{eff} was 1.00306 and the monitored K_{eff} was 1.00407, yielding a ΔK/K value of 0.1%. Thus, there is excellent agreement between the measured reactivity and cold prediction. A second reactivity anomaly test was performed at approximately 1150.5 MWD/MT (Megawatt Days/Metric Ton) of exposure and full power equilibrium xenon. This second test demonstrated excellent agreement with the hot reactivity prediction.

Control Rod Drive Functional Testing

After completion of fuel movements and subsequent core verifications, the initial timing of the normal insertion and withdrawal of each control rod was verified to be within the normal range of 40 to 60 seconds as derived from Nuclear Steam Supply System (NSSS) vendor recommendations. However, 8 control rods exhibited notching anomalies. The withdrawal times of these 8 rods were slowed, within specified limits, to correct the condition.

In accordance with WNP-2 Technical Specification 3/4.1.3.2, scram time measurements were performed on all control rods prior to exceeding 40% reactor power. Each control rod was individually scram time tested to Notch 6. The slowest measured scram time was 2.828 (Control Rod 26-23 to Notch 5); the acceptance criterion is ≤ 7 seconds.

The scram times for 4 control rods in a two-by-two array were also measured for compliance with Technical Specification 3/4.1.3.4 and the times assumed in the minimum critical power ratio (MCPR) analysis. These criteria are provided in Table 2 below:

Table 2: Average Scram Time Criteria for 4 Rods in Two-By-Two Array

Notch	Tech. Spec. Average Scram Time (seconds)	COLR Average Scram Time (seconds)
45	0.430	0.430
39	0.868	0.720
25	1.936	1.600
05	3.497	2.950

The measured scram times for the individual control rods were reviewed and all rods were faster than the Technical Specification limits at all notch positions. Hence, the results satisfied the Technical Specification average time criteria for 4 rod/two-by-two arrays. The measured scram times for the individual rods met the COLR acceptance criteria at all notch positions except one.

One control rod had a scram time of 0.721 seconds to Notch 39, compared to the COLR requirement of 0.720 seconds. However, the average time to Notch 39 was acceptable for the 4 rod/two-by-two array containing this rod. Pursuant to the requirements outlined in the COLR, if the average scram time of any 4 rods in a two-by-two array exceeds the COLR required times (as indicated in Table 2), the plant must operate with more conservative MCPR limits. Since the average scram times met the COLR requirements, no supplemental MCPR conservatism was necessary.

Core Power Symmetry Analysis

During the initial scram time testing prior to 40% power, Control Rod 50-19 became inoperable in the fully inserted position. This control rod is an out-of-sequence rod for the A2 control cell core configuration used in Cycle 12. Subsequent testing also found that Control Rod 54-47 would not settle into some notch positions. It was, therefore, inserted into the reactor core and declared inoperable. After achieving 100% reactor power on September 5, 1996, TIP measurements were performed with a non-symmetric rod pattern including Control Rods 50-19 and 54-47. An analysis of the TIP test data with reactor power near 100% showed that the TIP system met the statistical criterion for symmetry as established in Reactor Analysis Procedure

PPM 9.3.5, "TIP Symmetry Analysis." The statistical criterion is a CHI-Squared (X^2) test of significance, with the significance level fixed at 1% and requiring a X^2 value of less than 36.19. The analysis of the actual data resulted in a X^2 value of 17.106. Typically, the X^2 value is less than 5 with all control rods operable. Review of the individual data related to the asymmetric control rod pattern revealed that the actual asymmetries were as expected. Moreover, a calculated symmetry analysis, which is insensitive to asymmetrical control rod patterns, determined that the TIP symmetry results were acceptable and were similar to the results from previous operating cycles.

LPRM and TIP Response to Control Rod Motion

Individual LPRM response during control rod motion was checked during control rod functional testing to verify that the LPRMs were connected correctly and functioning properly. The TIP systems were verified to be connected and configured correctly by observing their response to control rod motion at each of the available core locations. Proper response from the operable LPRMs and all of the TIPs was verified.

Attachment 2

DFW/ASD Startup and Power Ascension Report

ASD/DFW MODIFICATIONS

As discussed in the cover letter, the R-11 outage scope included installation of new DFW and RRC pump ASD control systems. Both of these modifications were inspected by the NRC in June 1996 during installation and pre-operational testing (Reference 2).

The DFW upgrade modification replaced the existing Bailey 7000 series reactor feedwater (RFW) analog control system with a GE-FANUC 90-70 microprocessor based control system and the single unit electro-hydraulic actuator system for the RFW pump steam admission valves was replaced with a redundant Lovejoy digital control system. The DFW modification also included replacing the limit switches on the RFW pump minimum flow recirc valve with a position transmitter and upgrading of fuel zone level indication. The new DFW system continues to utilize the single element (reactor level) and three element (steam flow, feed flow, and reactor level) control modes for reactor vessel level control.

The ASD upgrade modification replaced the existing RRC electro-hydraulic flow control valve based flow control system and two-speed drive system for the RRC pumps (using motor-generator sets for slow speed operation) with a solid state variable speed ASD system. The ASD modification included deactivating the motor-generator sets, permanently blocking open the flow control valves, removing the associated hydraulic lines and capping the penetrations, and removing the hydraulic line containment isolation valves. A further description of the ASD modification is contained in the Reference 1 Technical Specification Amendment Request.

Testing activities for the DFW and ASD control system modifications were integrated into a single Power Ascension Test and are therefore discussed together. First the Power Ascension Test details will be discussed, followed by a discussion of the results to determine impacts on responses important to safety. Testing was performed at various Test Conditions (TC) from initial heatup through full power to verify that the Level 1 and 2 criteria that were met during initial plant startup are still satisfied.

Power Ascension Test Report Details

A) Reactor Feedwater Startup Flow Control Valve Stability Testing (TC Heatup)

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the startup flow control valve. The test was performed at 2% power with the Startup Level Control Valve, RFW-FCV-10, controlling water flow to the vessel and its controller, RFW-LIC-620, in AUTO with greater than or equal to 5% DEMAND.

Results

In all cases, the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. Evaluation of the Transient Data Acquisition System (TDAS) plot

data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. RFW-FCV-10 responded as designed to maintain reactor vessel water level following each step transient.

B) **Reactor Feedwater Startup Flow Control Valve Stability Testing (TC-1)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the startup flow control valve. The test was performed at 13% power with the Startup Level Control Valve, RFW-FCV-10, controlling water flow to the vessel and its controller, RFW-LIC-620, in AUTO with greater than or equal to 5% DEMAND.

Results

In all cases the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. Evaluation of the TDAS plot data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. RFW-FCV-10 responded as designed to maintain reactor vessel water level following each step transient.

C) **Harmonic/Vibration Testing**

Description of Test

This section ensured that the proper data were collected to evaluate electric bus harmonics and piping vibration. The data were gathered by Supply System test coordinators and GE contractors at various levels between initial heatup through 100% power.

Results

Analysis of the harmonics data has been completed and the results indicate that the harmonics are satisfactory. In addition, similar analysis has been completed for piping vibration studies and preliminary results indicate satisfactory acceptable vibration levels in most cases. Modifications to some piping systems may be required during the next refueling outage in order to achieve long term reliable operation.

D) **RFW-LIC-600 Single Element Level Step Changes (TC-2)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the DFW system. The test was performed at 26.5% power with the Reactor Presure Vessel (RPV) Level Master Controller, RFW-LIC-600, in AUTO and in Single Element control.

Results

In all cases the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. Evaluation of the TDAS plot data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. The DFW system, while in Single Element control, responded as designed to maintain reactor vessel water level following each step transient.

E) **RFW-LIC-600 Three Element Level Step Changes (TC-2)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the DFW system. The test was performed at 28.6% power with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control.

Results

In all cases the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. Evaluation of the TDAS plot data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. The DFW system, while in Three Element control, responded as designed to maintain reactor vessel water level following each step transient.

F) **RFW-LIC-600 Three Element Step Changes (TC-3)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the DFW system. The test was performed at 60% power with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control.

Results

In all cases the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. Evaluation of the TDAS plot data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. The DFW system, while in Three Element control, responded as designed to maintain reactor vessel water level following each step transient.

G) **Manual Flow Steps - Two RFW Pump Operation (TC-3)**

Description of Test

This test initiated +3% (≈180 RPM) and -3%, then +8% (≈480 RPM) and -8% transient manual flow step changes to observe the performance of the DFW system. The test was performed at 64% power with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control and both Turbine Speed Controllers, RFW-SC-601A and 601B, in AUTO.

Results

In all cases the transient response of the system did not diverge, meeting Level 1 RFW Acceptance Criteria. During both 3% and 8% step changes, the Open Loop response did not meet the Level 2 Acceptance criteria for Delay Time, Response Time, Settling Time and Percent Overshoot. The system was determined operable and testing was allowed to continue. A decision was made to continue testing based on the fact that optimum tune conditions and criteria specifications were written for 100% operation. Tuning changes to improve the system performance at this lower power level were being prepared by Lovejoy Controls and GE.

Upon completion of the tuning changes, the 3% and 8% step changes at TC-3 conditions, open loop response Level 2 Acceptance Criteria were met. Tuning changes improved response time to enable the system to meet all of the Level 2 Acceptance Criteria.

H) **Reactor Recirculation Control System Speed Step Changes (TC-3)**

Description of Test

This test initiated 5% (90 RPM) speed changes in each of the RRC pumps under Individual control and then initiated a 1% (18 RPM) speed change while the pumps were in Master control to observe transient response. The test was performed with reactor power at 65% and the frequency of both ASDs started at approximately 45 Hz.

Results

In all cases the transient response of the system did not diverge, meeting the Level 1 RRC Acceptance Criteria. Evaluation of the TDAS plot data verified that the Level 2 RFW Acceptance Criteria were met with a system response decay ratio of less than 0.25. The ASDs adjusted speed and flow by evenly ramping to the new speed and flow, with no overshoot, and a solid stable response of the system.

I) **RRC-Single Drive Load Tests (TC-3)**

Description of Test

This test determined the maximum drive load setting of the ASD that placed the motors at 5600 HP. The test was performed at 64.4% power with the DFW system in AUTO and in Three Element control and both Feedwater Turbine Controllers were in AUTO. The frequency of both ASDs started at approximately 45 Hz and each ASD drive was switched from 12-pulse (two-channel) to 6-pulse (single-channel) operation.

Results

Testing found that the A1 drive channel reached 5600 HP at 51.2 Hz and the B1 drive channel reached 5600 HP at 51.7 Hz. Both of these drive channels then had their maximum 6-pulse speed limiter settings conservatively set at 51.0 Hz. The A2 and B2 drive channel 6-pulse speed limiters were also set at 51.0 Hz, based on common design and equivalent components.

J) **RRC Pump Maximum Speed Limits (TC-3)**

Description of Test

This test provided verification that the ASD and FANUC overspeed (overfrequency) settings are appropriately set to ensure that RRC pump speeds do not exceed 105% of their designed rating of 1782 RPM (1871 RPM). The test was performed at 63% power with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control. All of the drives where run up to each of the separate speed limit clamps of the ASD drive system and ASD FANUC system. The speed limiters settings are 62 Hz for the ASD Limiter Setpoint and 61.85 Hz for the FANUC Limiter Setpoint.

Results

The system responded as designed. Both of the Loops (A and B) were properly speed limited by both the ASD FANUC system and ASD Drive system.

RRC Pump A FANUC limited speed to 1867 RPM.
RRC Pump B FANUC limited speed to 1867 RPM.

RRC Pump A ASD Drive limited speed to 1865 RPM.
RRC Pump B ASD Drive limited speed to 1865 RPM.

These results are conservative with respect to the licensing and design bases.

K) Initiation of RRC System "RUNBACK" (TC-3)

Description of Test

This test was performed at 54% power and 62% rod line with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control. With the RRC pumps in Master (ganged) control, a runback signal was generated with the ASD FANUC computer to allow validation of average rate of change of RRC pump speed in percent per second. This test was conducted to verify Technical Specification 4.4.1.1.3 rate criteria.

This test also provided data to evaluate the capability of the DFW system to control water level such that there is sufficient margin to SCRAM. Control system response was evaluated for impact on reactor water level, total feedwater flow, reactor pressure, recirculation loop flows, total core flow, and one APRM channel.

Results

ASD frequency decreased to 27 Hz as designed and reactor power decreased from 54% to 32%. The DFW system controlled the water level from an initial 37 inches to a maximum of 41 inches during the runback. The digital feedwater system then brought water level back to 37 inches. The Acceptance Criterion was met that required reactor water level to be controlled to avoid a low water level scram (at 13 inches) by at least a three inch margin.

The following data were collected during the runback test.

Pump	Beginning Speed	Ending Speed	Calculated Rate	T. S. Criteria
RRC-P-1A	1253 RPM	807.6 RPM	4.95%/Sec	≤ 10%/Sec
RRC-P-1B	1229.4 RPM	809.1 RPM	4.67%/Sec	≤ 10%/Sec

An oscillatory response in water level control was observed following the 27 Hz runback. It was later determined that the two feedwater pumps were bumping against their minimum speed setpoints and were at the low end of automatic control. The system was evaluated and determined to be within its capability to maintain reactor water level with no additional burden placed on the operators.

L) **RFW-LIC-600 Three Element Step Changes (TC-5)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the DFW system. The test was performed at 71% power and a 97.8% rod line with the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control.

Results

In all cases the transient response of the system did not diverge and met all Level 1 and Level 2 RFW Acceptance Criteria.

M) **RFW-LIC-600 Three Element Step Changes (TC-6)**

Description of Test

This test initiated +3 inch and -3 inch, then +6 inch and -6 inch step changes to observe the performance of the DFW System. The test was performed at 96% power with RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control. In addition, the RPV water level mismatch between Single and Three Element control was tested.

Results

In all cases the transient response of the system did not diverge and met all Level 1 and Level 2 RFW Acceptance Criteria.

When the system was transferred from Three Element control to Single Element control a difference of 0.7 inch was observed which met the Level 2 Acceptance Criteria of within +/-1 inch.

This section was repeated a number of times in conjunction with tuning changes. Table 2.1 below contains the results of the feedwater turbine open-loop transient testing performed after tuning the DFW control system and provides a comparison of similar test results achieved for the original analog Feedwater control system as part of the 1984 (original) startup test program.

Table 2.1 - Feedwater Pump Turbine Open Loop Flow Response Summary

Feedwater Pump		Test Condition	Initial Flow (GPM)	Step Size (%)	Delay Time (Sec)	Response Time (Sec)	Over-shoot (%)	Settling Time (Sec)
A	Original S/U	TC-6	16000	+20	0.95	0.6	14.75	8.4
	Original S/U	TC-6	15000	+10	0.6	0.62	10.52	9.4
	1996	TC-6	14000	+8	0.8	1.6	0	6.2
	1996	TC-6	14000	-8	0.9	1.9	0	6
B	Original S/U	TC-6	15000	+20	0.7	0.65	15	6
	Original S/U	TC-6	15000	+10	0.6	0.8	10.41	9.2
	1996	TC-6	15000	+8	1	1.8	9.1	11
	1996	TC-6	15000	-8	1	1.7	15	12
	Criteria				1.1	1.9	15%	14 Sec

Analyses of the Reactor Feedwater Pump Turbine (RFPT) open loop transient response reveals the following:

- Following final software corrections and tuning of the Lovejoy Governor Control System, the A and B RFPT open loop transient response passed all Level 1 and Level 2 Acceptance Criteria.

- RFPT-A transient response is better than RFPT-B. This was due to the fact that the RFPT-A transient test was conducted with its auxiliary oil pump running, whereas the transient test for the RFPT-B was conducted with its auxiliary oil pump off. This contributes to a slower response from RFPT-B compared to RFPT-A for the same step change transient.

During Power Ascension Testing, the auxiliary oil pump for the A and B RFPTs were, at times, found to automatically start when the RFPT transitioned from the Low to the High Pressure Control Valve and during some +/-8% speed transient tests. The cause of the auxiliary oil pump auto start is due to the faster response of the post modification RFPT control system. The transient demand on the RFPT control oil system has increased, therefore a larger surge/makeup capacity during transients is desired. Although the RFPT control system met all listed Acceptance Criteria for transient response, it is desired to optimize the control oil system to match the enhanced GE/Lovejoy control system. Therefore, a RFPT Control Oil Performance Evaluation will be performed during the next refueling outage to critically evaluate the hydraulic performance of the system and determine modifications as appropriate.

- The "overshoot" (the percentage above the final demanded speed by which the RFPT speed rises before converging on the setpoint) characteristic of the new Lovejoy Governor control system is significantly smaller than that of the previous control system.

  The Lovejoy control system response to a step change between 10% and 95% uses proportional integration, with the remaining 5% of the step broken into several smaller proportional steps. This method is called "Isochronous Corrections" and provides for a system response with less overshoot. Although this causes a slightly larger "response time" (the time it takes for the RFPT speed to transition from 10% to 90% of the final setpoint, compared to the original setpoint), this control scheme is desirable since it reduces the overall transient time of the system, thereby providing a shorter time to achieve a new steady state level following a transient.

The following three figures provide a comparison between the analog Feedwater control system response in 1984 (initial) PAT and the DFW control system PAT performed in 1996. Figure 2.1 compares the six inch step changes at approximately 95% power. Figures 2.2 and 2.3 show the response of each feedwater turbine to the six inch level step change. In the figures, the results for the post-R-11 PAT are labeled "DFWLC [Digital Feedwater Level Control]" and the test results for the intial PAT are labeled "AFWLC [Analog Feedwater Level Control]."

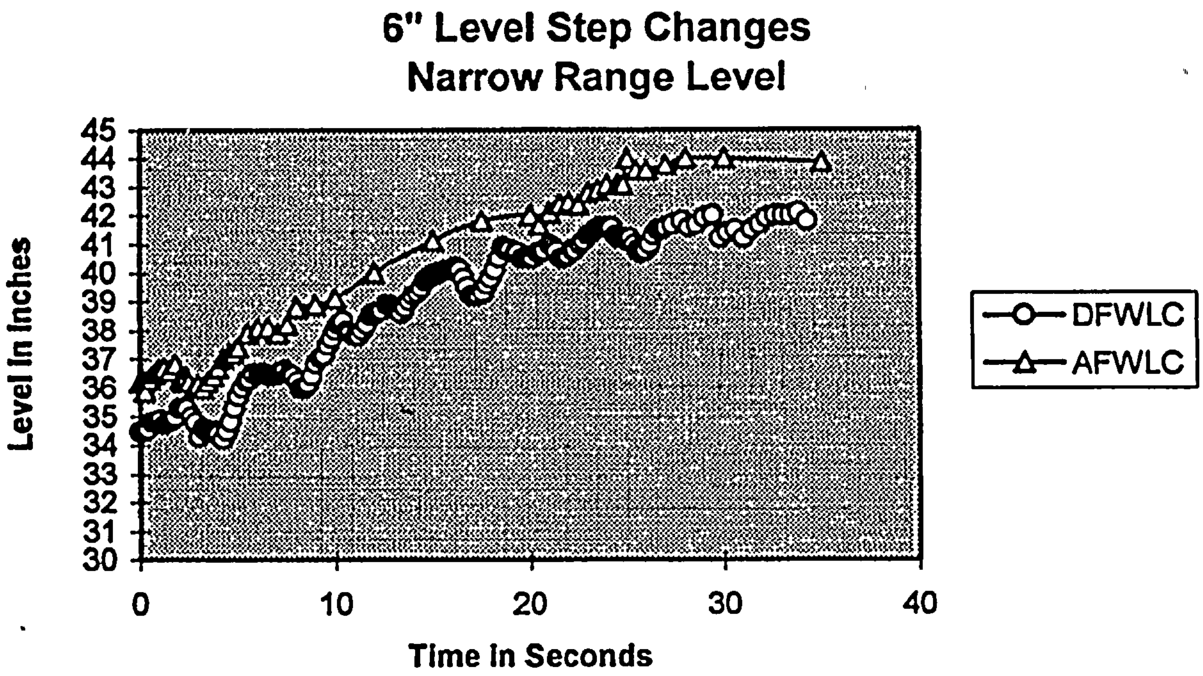

Figure 2.1 - Six Inch Level Step Change

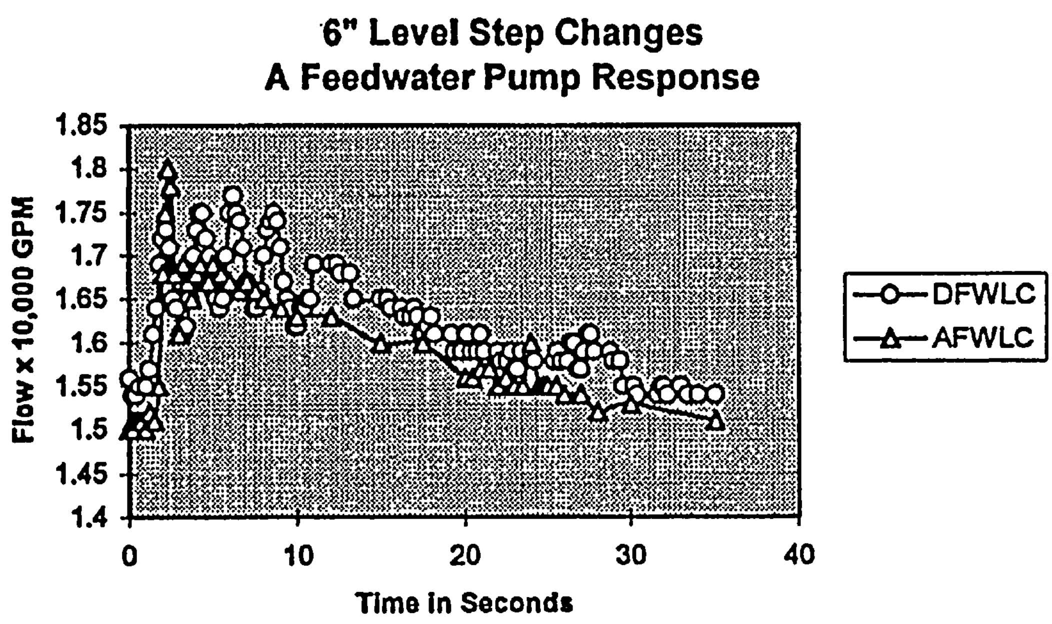

Figure 2.2 - A Feedwater Pump Response to Six Inch Level Step Change

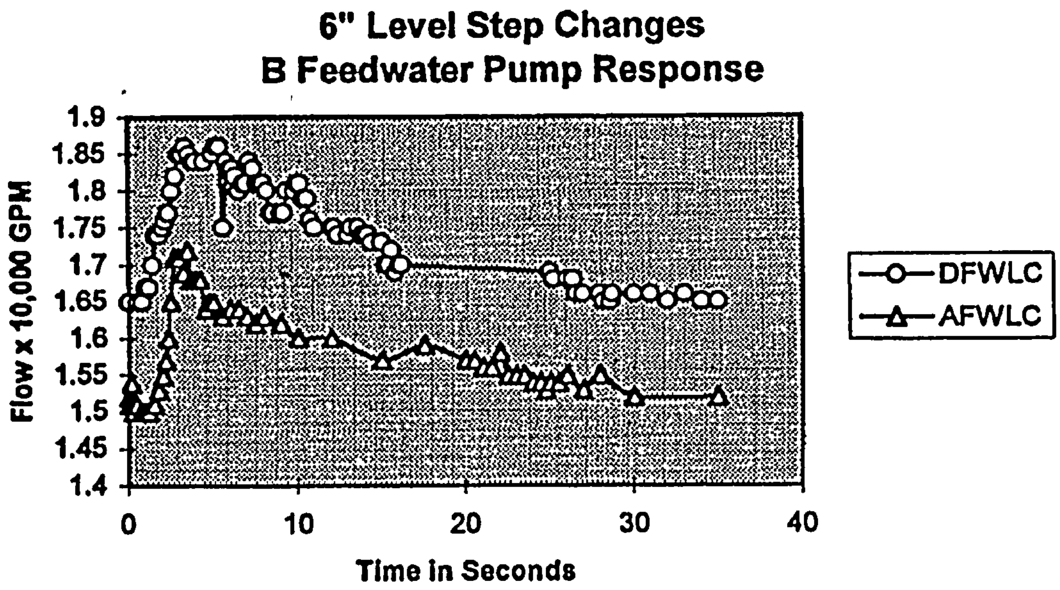

Figure 2.3 - B Feedwater Pump Response to Six Inch Level Step Change

N) RRC Pump Maximum Speed Limits

Description of Test

This test was performed to confirm that the maximum core flow achievable would not be greater that 108.5% of design. With plant power at 97.4% power and the RPV Level Master Controller, RFW-LIC-600, in AUTO and in Three Element control, the test verified the ASD and Recirculation Pump Trip (RPT) breaker overspeed (overfrequency) settings by extrapolation of data as the RRC system operated at its maximum speed limits and core flow.

Results

Testing and use of extrapolation verified that an ASD frequency of 65.4 Hz corresponds to 108.5% core flow. Therefore, the overfrequency relay setting of 64.7 Hz will conservatively protect the ASD from exceeding 108.5% core flow.

Conclusion

The performance of the Power Ascension Test has proven that the installation of the ASD and DFW systems meets all design basis system performance criteria. Comparison of the new DFW control system versus the previous analog Feedwater control system responses reveals that the new system controls reactor vessel water level as well as the previous system for similar tests. A number of issues which impact the performance of the DFW and ASD control systems were uncovered as a result of the PAT process and post-R-11 outage startup testing/operation. Although many issues have been resolved to date, several issues remain outstanding and must be resolved to assure optimum system performance. However, the DFW and ASD systems continue to perform their design basis functions with the outstanding items.

EVALUATION OF IMPACTS ON IMPORTANT TO SAFETY RESPONSES

This analysis is focused on criteria that provide assurance of the safety of plant operation. Specifically, the plant response to accidents and Abnormal Operational Occurrences (AOOs) analyzed in reload analyses are required to meet the WNP-2 Final Safety Analysis (FSAR) design bases and assumptions.

Comparison of System Responses with Initial Power Ascension Test

The water level setpoint 6 inch step change test at TC6 (in this case 96% rated power) is used in the comparison. Selection of this test is based on the fact that a key parameter in the limiting transients for WNP-2 (Generator Load Rejection for thermal limits and Main Steam Isolation Valve Closure for the ASME overpressure limit) is the water level response. Water level influences the rate of pressurization of the limiting transient and the magnitude of the power excursion during the transient.

The final 6 inch level step change test for the DFW system was performed on October 13, 1996. The narrow range water level and the Pump A and B flows were ploted as a function of time as shown in Figures 2.1 through 2.3. The results for the post-R-11 PAT are labeled "DFWLC [Digital Feedwater Level Control]." In the figures, the test results for the same test during the initial PAT, performed on December 1, 1984, are also provided and labeled "AFWLC [Analog Feedwater Level Control]". It is seen from Figure 2.1 that the DFW response closely follows that from the PAT. The data from the DFW test has an error band of about +/-0.5 inch. The initial PAT test data has a smoother curve because of lower resolution. If one allows the same error band of +/-0.5 inch for the initial PAT test, and ignores the initial offset (which is not material in this case), the two curves would overlap. The offset between the two curves is due to the different starting points at the initiation of the transients. From the above discussion, Figure 2.1 indicates that the water level change rate is essentially the same as that from the initial PAT.

Figure 2.2 compares the Pump A flow after the level setpoint change. It is seen that the initial flow for the DFW control system is higher. This is due to higher feedwater flow required for power uprate which was implemented in 1995. The initial PAT was tested at a power level of

95% of 3323 MWt. The post-R-11 DFW test was performed at 96% of 3486 MWt. The initial overshoot of the pump flow is somewhat different from the initial PAT result, but the decay of the flow is very similar to the initial PAT result. If the two curves are normalized, they would be very close to each other during the decay phase. The initial oscillation of the flow was caused by the oscillatory behavior of RFW Governor Valve A. This oscillatory response does not impact the transient analysis since the determining factor is the water level response. Although the oscillation does not affect normal plant operation, the issue will be resolved through a future plant modification. Figure 2.3 gives Feed Pump B flow comparison. As with Pump A, the flow responses for both the DFW control system and analog Feedwater control system are similar. The offset between the two curves are caused by two factors. The first is due to power uprate and the second is due to the pump speed mismatch between Pump A and Pump B. Pump B has been biased to run at higher speed than Pump A to minimize pump vibration. In the initial PAT, both pumps were running at the same speed.

From the comparison of the responses between the DFW and analog Feedwater control systems for the level step change test, one can conclude that the digital system has been tuned to provide essentially the same response for parameters important to transient analysis. It also leads to the conclusion that the transient models approved by the NRC and used by the Supply System and the fuel supplier (ABB) remain valid after the DFW modification.

Sensitivity of Feedwater Flow on Limiting Transient Analysis

To provide further justification for the conclusion that the DFW control system does not change the result of the limiting transient analysis, a comparison of the water levels for the generator load rejection without bypass (LRNB) transient for two different feedwater flows was performed. One case assumed the feedwater flow during the LRNB transient does not change. The second case used a feedwater flow as calculated by a DFW control system model. The constant flow case is the model used by ABB (BISON model). The variable flow case is the model used by the Supply System (RETRAN model). The calculated water level as a function of time for the same LRNB transient at End-of-Cycle 12 (EOC12) was compared with the Technical Specification Scram Time with the RPT operable. The water level between the two models compared very well from initiation to about 1.25 seconds. After 1.25 seconds the ABB model gives higher water level than the RETRAN model. This is due to the higher feed flow assumed by the ABB model. Even though there is a divergence of the water level after 1.25 seconds, the resulting thermal limits from these calculations are not impacted by the difference because the lowest MCPR occurs at about 1.2 seconds.

The resulting water level up to the point of lowest thermal limit was determined to be insensitive to the feedwater flow model input in a limiting transient. This results from the fact that the determining parameter in a limiting transient is vessel pressure, not feedwater flow. This supports the conclusion that the DFW/ASD and analog Feedwater control system responses do not result in different operating limits for the limiting transients.

Sensitivity of Recirculation Flow Response on Limiting Transient Analysis

The impact of the ASD control on the limiting transients is insignificant since all limiting pressurization transients (Load Rejection, Turbine Trip, Feedwater Controller Failure, and MSIV Closure) use the EOC-RPT or Anticipated Transient Without Scram (ATWS)-RPT as the mitigating trip almost immediately following initiation of the transients. EOC-RPT and ATWS-RPT are functions independent from ASD control. The RRC pump coastdown characteristics following an RPT trip have not changed as a result of ASD implementation. Therefore, the ASD response does not impact the limiting transient calculations.

In the case where the EOC-RPT function is bypassed, recirculation flow is assumed to be constant in the transient analyses and an additional MCPR penalty is applied as prescribed in the COLR.

Other FSAR Chapter 15 Transients

Other transients that could be impacted by DFW/ASD control system modification are: 1) Feedwater Controller Failure - Maximum Demand (FWCF); 2) Loss of Feedwater Flow (LOFW); 3) Recirculation Pump Trip (RPT); 4) Startup of Idle Recirculation Pump (SIRP); 5) Recirculation Flow Control Failure with Increasing Flow (RFCFI); and 6) Recirculation Flow Control Failure with Decreasing Flow (RFCFD).

For the FWCF transient, the feedwater flow is conservatively assumed to instantaneously increase to the maximum value of 139% of rated. The outcome of the analysis of this transient is not affected by the DFW/ASD responses because the Feedwater system is designed to not exceed 139% of rated flow even under failure mode.

The LOFW transient involves a reduction of feedwater flow; therefore, there will be no power excursion after the initiation of the transient. The LOFW transient is bounded by the LRNB transient. The DFW/ASD responses will not impact the calculated safety limits for the LOFW transient and therefore is not affected by the DFW/ASD modifications.

The RPT transient is also bounded by the LRNB transient since there is no power excursion involved with this transient. ASD implementation has no impact on the RRC pump coastdown characteristics following an RPT transient.

The SIRP transient is based on the assumption that the restart of an idle pump results in the RRC pump achieving rated speed. The bounding assumptions used in evaluating this transient are not affected by DFW/ASD modifications. RRC pump starts with the ASD are "soft" starts which increase safety margins. Testing done during the initial PAT is therefore bounding. Since a SIRP involves operator manual action and since this transient is not a limiting transient, the DFW/ASD response does not impact the safety limit calculation.

For the RFCFI transient, the Cycle 12 reload analysis takes into consideration the higher maximum flow as set by the high speed limiter in the ASD. The ASD modification is reflected in the flow-dependent MCPR limits in the Cycle 12 COLR. The ASD high speed limiters were set and verified correct based on plant testing. The bounding assumptions used in evaluating this transient are not affected by the DFW/ASD changes.

The RFCFD is a power-reduction transient and is not a limiting transient. The DFW/ASD responses will not impact the safety limits for this transient.

Conclusion

Based on the comparison of the level step change test data from the post-DFW/ASD implementation PAT and the initial PAT, it is concluded that the important to safety considerations are not changed by implementation of the DFW/ASD control systems. In addition, an evaluation of the impact on the FSAR Chapter 15 transients was performed and testing demonstrates that the DFW/ASD control system responses have no adverse impact on the Chapter 15 transient analyses. The assumptions used in the transient analyses bound the initial PAT results and current testing shows that the DFW/ASD control system response replicates the initial PAT response.