COMMONWEALTH EDISON COMPANY

BRAIDWOOD UNIT 1 CYCLE 1

STARTUP REPORT

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1.0 INTRODUCTION

This report describes the required testing at Braidwood Station Unit 1 from the time the first fuel assembly was loaded into the reactor until the plant achieved initial criticality. It satisfies the requirement of the Braidwood Unit 1 Technical Specifications that a Startup Report be submitted to the NRC nine (9) months following initial criticality.

Braidwood Station, located in northeastern Illinois, utilizes a four loop Westinghouse pressurized water reactor system. Westinghouse Electric Corporation, Sargent & Lundy, and the Commonwealth Edison Company jointly participated in the design and construction of Braidwood Unit 1. The plant is operated by Commonwealth Edison Company with Sargent & Lundy as the Architect - Engineer.

The nuclear steam supply system is designed for a power output of 3411 MWt. The equivalent warranted gross and approximate net electrical output are 1175 MWe and 1120 MWe, respectively. Cooling for the plant is provided by a large man-made cooling pond of approximately 2500 acres constructed over a previously strip-mined area. Essential service cooling is provided by a 99-acre auxiliary cooling pond which is integral with the main pond.

2.0 DISCUSSION OF BRAIDWOOD STARTUP PROGRAM

The Braidwood Unit 1 startup testing program consisted of single and multisystem tests that occurred commencing with initial fuel loading and will continue through full power. These tests demonstrate overall plant performance and include such activities as precritical testing, low power tests, and power ascension tests. Testing sequence documents are utilized for each plateau to coordinate the sequence of testing activities at that plateau.

In the subsections that follow, a description of the sequence of testing at each plateau is provided. Also included as a part of Section 2.0 is a table showing major milestones for Braidwood Unit 1 which occurred during the startup program, prior to initial criticality, and a list of operational modes as defined by the Technical Specifications.

TABLE 2.0 - 1

BRAIDWOOD UNIT 1 MAJOR MILESTONES

MAJOR MILESTONES	DATE
Precritical License Issued	10/17/87
Fuel Load Commenced	10/25/87
5% Power License Issued	05/21/87
Initial Criticality	05/29/87

TABLE 2.0 - 2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

2.1 INITIAL CORE LOAD SEQUENCE

The Core Load Sequence Document was utilized to coordinate the sequence of operations associated with the initial core loading program. This sequence included scheduling of the individual startup tests associated with core loading. This document specified as prerequisites which testing had to be completed prior to commencement of core loading, the required status of the plant systems necessary to support core loading, and the reactor vessel status. A log was also included in the sequence document to verify containment evacuation alarm operability throughout core loading. This document also provided the criteria for stopping core loading and the actions to be followed prior to the resumption of core loading, in the event core loading was stopped prior to completion. Results of individual tests completed during the core loading sequence are discussed in Section 3.1 of this report. Upon completion of core loading, plant systems were aligned as directed by the Shift Engineer.

2.2 POST CORE LOADING PRECRITICAL TEST SEQUENCE (PCLPC)

The PCLPC Sequence Document was utilized to define the sequence of tests and operations to be performed between completion of initial core loading and prior to initial criticality. This document ensured that core load testing had been successfully completed and results approved prior to continuation of the testing program. This document scheduled the performance of precritical tests to ensure the necessary testing was completed prior to initial criticality. Plant operating procedures were utilized where appropriate to establish necessary plant conditions. Results of individual tests completed during the post core load precritical testing phase are discussed primarily in Section 3.2 of this report. Upon completion of this testing phase, plant systems were restored as directed by the Shift Engineer.

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3.1 CORE LOADING

3.1.1 - INITIAL CORE LOADING FH-33

OBJECTIVES

The Initial Core Loading Test Procedure was performed to ensure that the nuclear fuel assemblies were loaded in a safe and cautious manner such that an inadvertent criticality was avoided. This procedure was also utilized to verify placement of the fuel assemblies into their proper core locations upon completion of core load.

TEST METHODOLOGY

The test procedure began by loading the temporary core loading instrumentation into their initial positions and determining background count rates for all source range and temporary nuclear instrumentation channels. The two primary source bearing assemblies and six additional assemblies, comprising the 'source nucleus', were loaded next. Audible indication of neutron population changes from one of the two installed plant channels, was maintained in both the control room and containment for the duration of the core loading process. After the source nucleus assemblies were loaded, the average of ten sets of counts measured over a 100 second time period was taken for all five nuclear channels used in the core loading process (two source range and three temporary channels). The first reference value, for use in inverse count rate ratio monitoring, was determined from these sets of counts after appropriate background values had been subtracted. Subsequent reference values were calculated whenever core loading was suspended for eight hours or longer, a temporary detector was moved, or a primary source bearing assembly was moved to a different core location.

Inverse count rate ratio monitoring was used following each fuel assembly move to ensure that the reactor was not approaching criticality. To ensure reliability in the monitoring, a minimum of two of the five nuclear instrumentation channels were required to be responding to source neutron population changes throughout core loading. Data obtained during inverse count rate ratio monitoring was compared to actual data supplied by Westinghouse Electric Corporation from the Salem Unit Two core loading. Salem Unit Two data, having a similar core loading pattern to that used at Braidwood, allowed core loading personnel to verify that the core was being loaded in a safe and cautious manner.

Upon completion of loading, the core was mapped using a video camera to verify proper placement of assemblies into the reactor vessel.

3.1.1 - INITIAL CORE LOADING, FH-33 (Continued)

SUMMARY OF RESULTS

Core loading was completed in a safe and cautious manner as required by the acceptance criteria of the core loading procedure. Problems encountered during the test were primarly associated with non-responding or malfunctioning neutron detectors and the Fuel Handling System.

Noise spiking problems on the two installed source range channels caused several delays during fuel load. The observed spikes were of varying magnitudes and durations, and most resulted in actuation of the containment evaulation alarm. Several of the spikes were determined to be caused by construction activities (welding) near the cable penetrations or operating related activities in the switchyard.

Neutron detector malfunctions were attributed to leakage into the detectors due to degraded O-rings and out of round detector cans.

Several problems arose with the Fuel Handling System. A shear pin on the drive system sheared due to misalignment of the drive mechanism. The pin was replaced and the mechanism adjusted. The Fuel Handling Building upender latch mechanism became bent due to galling of the latch bushing. A grease plug was discovered missing on the manipulator crane. A new plug was installed and an exhaustive search was performed to assure the plug had not fallen into the reactor cavity. All these problems resulted in delays with core loading but did not directly affect the loading procedure.

FIGURE 3.1.1-1

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C		C30S	C46S	C48S						
2				B			C35S	A27S	• C37S			A			
3															
4															
5															
6															
7															
8															
9															
10															
11															C44S
12															
13															
14													• C57S		
15															

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #1

FIGURE 3.1.1-2

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C30S	C46S	C48S	C03S					
2				B			C35S	A27S	• C37S	A30S		A			
3							A49S	B30S	A03S	B46S					
4							B12S	A42S	B39S	A35S					
5							A29S	B01S	A01S	B45S					
6							B43S	A52S	B64S	A44S					
7							A13S	B55S	A56S	B11S					
8						C	B52S	A55S	A54S	A64S					
9															
10															
11															C44S
12															
13															
14													• C57S		
15															

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #2

FIGURE 3.1.1-3

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C30S	C46S	C48S	C03S					
2				B			C35S	A27S	•C37S	A30S		A			
3							A49S	B30S	A03S	B46S					
4							B12S	A42S	B39S	A35S					
5							A29S	B01S	A01S	B45S					
6							B43S	A52S	B64S	A44S					
7							A13S	B55S	A56S	B11S					
8						C	B52S	A55S	B54S	A64S					
9							A10S	B19S	A20S	B10S					
10							B15S	A57S	B47S	A34S					
11							A08S	B09S	A17S	B08S					
12							B04S	A04S	B02S	A02S					
13							A59S	B22S	A61S	B21S					
14			C44S						•C57S	A37S					
15															

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #3

FIGURE 3.1.1-4

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C30S	C46S	C48S	C03S	C59S				
2				B			C35S	A27S	•C37S	A30S	C26S				
3							A49S	B30S	A03S	B46S	A60S				
4							B12S	A42S	B39S	A35S	B41S				
5							A29S	B01S	A01S	B45S	B24S				
6							B43S	A52S	B64S	A44S	B42S				
7							A13S	B55S	A56S	B11S	A11S	B28S	A28S	C49S	C33S
8						C	B52S	A55S	B54S	A64S	B63S	A06S	B06S	A63S	A
9							A10S	B19S	A20S	B10S	A32S	B20S	A31S	C62S	C07S
10							B15S	A57S	B47S	A34S	B36S				
11							A08S	B09S	A17S	B08S	B17S				
12							B04S	A04S	B02S	A02S	B23S				
13							A59S	B22S	A61S	B21S	A22S				
14							C44S	A41S	•C57S	A37S	C18S				
15							C58S	C60S	C22S	C21S	C40S				

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #4

FIGURE 3.1.1-5

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							C30S	C46S	C48S	C03S	C59S				
2				B			C35S	A27S	•C37S	A30S	C26S	C29S	C01S		
3							A49S	B30S	A03S	B46S	A60S	B48S	C14S	C53S	
4							B12S	A42S	B39S	A35S	B41S	A38S	B35S	C43S	
5							A29S	B01S	A01S	B45S	B24S	B14S	A46S	C38S	C50S
6							B43S	A52S	B64S	A44S	B42S	A65S	B38S	A43S	C55S
7							A13S	B55S	A56S	B11S	A11S	B28S	A28S	C49S	C33S
8							B52S	A55S	B54S	A64S	B63S	A06S	B06S	A63S	A
9							A10S	B19S	A20S	B10S	A32S	B20S	A31S	C62S	C07S
10							B15S	A57S	B47S	A34S	B36S	A16S	B56S	A25S	C10S
11							A08S	B09S	A17S	B08S	B17S	B44S	A45S	C31S	C15S
12							B04S	A04S	B02S	A02S	B23S	A23S	B61S	C36S	
13							A59S	B22S	A61S	B21S	A22S	B40S	C16S	C25S	
14							C44S	A41S	•C57S	A37S	C18S	C04S	C02S		
15					C		C58S	C60S	C22S	C21S	C40S				

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #5

FIGURE 3.1.1-6

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C05S	C27S	C30S	C46S	C48S	C03S	C59S				
2					C41S	A05S	C35S	A27S	• C37S	A30S	C26S	C29S	C01S		
3					A47S	B27S	A49S	B30S	A03S	B46S	A60S	B48S	C14S	C53S	
4					B59S	A40S	B12S	A42S	B39S	A35S	B41S	A38S	B35S	C43S	
5					B26S	B29S	A29S	B01S	A01S	B45S	B24S	B14S	A46S	C38S	C50S
6					B51S	A54S	B43S	A52S	B64S	A44S	B42S	A65S	B38S	A43S	C55S
7	C11S	C28S	A39S	B50S	A51S	B13S	A13S	B55S	A56S	B11S	A11S	B28S	A28S	C49S	C33S
8	B	A50S	B49S	A33S	B33S	A53S	B52S	A55S	B54S	A64S	B63S	A06S	B06S	A63S	A
9	C63S	C06S	A07S	B62S	A19S	B07S	A10S	B19S	A20S	B10S	A32S	B20S	A31S	C62S	C07S
10					B31S	A36S	B15S	A57S	B47S	A34S	B36S	A16S	B56S	A25S	C10S
11					B16S	B25S	A08S	B09S	A17S	B08S	B17S	B44S	A45S	C31S	C15S
12					B18S	A18S	B04S	A04S	B02S	A02S	B23S	A23S	B61S	C36S	
13					A62S	B60S	A59S	B22S	A61S	B21S	A22S	B40S	C16S	C25S	
14					C17S	A21S	C44S	A41S	• C57S	A37S	C18S	C04S	C02S		
15					C	C61S	C58S	C60S	C22S	C21S	C40S				

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

CORE CONFIGURTION #6

FIGURE 3.1.1-7

BRAIDWOOD UNIT 1 CYCLE 1
RANDOMIZED CORE LOADING PATTERN
FINAL CONFIGURATION

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					C05S	C27S	C30S	C46S	C48S	C03S	C59S				
2			C12S	C39S	C41S	A05S	C35S	A27S	• C37S	A30S	C26S	C29S	C01S		
3		C45S	C24S	B05S	A47S	B27S	A49S	B30S	A03S	B46S	A60S	B48S	C14S	C53S	
4		C51S	B03S	A12S	B59S	A40S	B12S	A42S	B39S	A35S	B41S	A38S	B35S	C43S	
5	C64S	C42S	A26S	B37S	B26S	B29S	A29S	B01S	A01S	B45S	B24S	B14S	A46S	C38S	C50S
6	C13S	A24S	B53S	A14S	B51S	A54S	B43S	A52S	B64S	A44S	B42S	A65S	B38S	A43S	C55S
7	C11S	C28S	A39S	B50S	A51S	B13S	A13S	B55S	A56S	B11S	A11S	B28S	A28S	C49S	C33S
8	C52S	A50S	B49S	A33S	B33S	A53S	B52S	A55S	B54S	A64S	B63S	A06S	B06S	A63S	C54S
9	C63S	C06S	A07S	B62S	A19S	B07S	A10S	B19S	A20S	B10S	A32S	B20S	A31S	C62S	C07S
10	C19S	A15S	B32S	A48S	B31S	A36S	B15S	A57S	B47S	A34S	B36S	A16S	B56S	A25S	C10S
11	C20S	C32S	A09S	B34S	B16S	B25S	A08S	B09S	A17S	B08S	B17S	B44S	A45S	C31S	C15S
12		C47S	B57S	A58S	B18S	A18S	B04S	A04S	B02S	A02S	B23S	A23S	B61S	C36S	
13		C56S	C34S	B58S	A62S	B60S	A59S	B22S	A61S	B21S	A22S	B40S	C16S	C25S	
14			C09S	C08S	C17S	A21S	C44S	A41S	• C57S	A37S	C18S	C04S	C02S		
15					C23S	C61S	C58S	C60S	C22S	C21S	C40S				

0° (E)

Notes
1) The equipment hatch is at 90° inside Unit One Containment

2) • Denotes Primary Source Location

FINAL CORE CONFIGURTION

FIGURE 3.1.1-8

SHUTDOWN AND CONTROL ROD LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				S_A		B		C		B		S_A			
3					S_D		S_B		S_B		S_C				
4		S_A		D				S_E				D		S_A	
5			S_C										S_D		
6		B				C		A		C				B	
7			S_B										S_B		
8		C		S_E		A		D		A		S_E		C	
9			S_B										S_B		
10		B				C		A		C				B	
11			S_D										S_C		
12		S_A		D				S_E				D		S_A	
13					S_C		S_B		S_B		S_D				
14				S_A		B		C		B		S_A			
15															

0° (E)

CONTROL BANK	NUMBER OF RODS
A	4
B	8
C	8
D	5
TOTAL	25

SHUTDOWN BANK	NUMBER OF RODS
S_A	8
S_B	8
S_C	4
S_D	4
S_E	4
TOTAL	28

FIGURE 3.1.1-9

BURNABLE POISON ROD LOADING PATTERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							12A	6	12A						
2			5		20		24		23 1S		20		5		
3		5		24		24		20		24		24		5	
4			24		20	4S	24		24		20		24		
5		20		20	12	24		20		24	12	20			
6			24		24		24		24		24		24		
7	12A	24		24		24		24		24		24		24	12A
8	6		20		20		24		24		20		20		6
9	12A	24		24		24		24		24		24		24	12A
10			24		24		24		24		24		24		
11		20		20	12	24		20		24	12	20		20	
12			24		20		24		24	4S	20		24		
13		5		24		24		20		24		24		5	
14			5		20		24		23 1S		20		5		
15							12A	6	12A						

0° (E)

Number Indicates Number of Burnable Poison Rods

S Indicates Source Rod

1934 BP Rods

12.5 w/o B_2O_3

3.1.2.a - CORE LOADING INSTRUMENTATION (HIGH VOLTAGE/DISCRIMINATOR/ NEUTRON CHECK), NR-30A

OBJECTIVE

The core loading instrumentation test procedure was performed prior to core loading to determine the proper operating and discriminator voltage settings for the temporary core loading instrumentation and to verify that the instruments provided indication when a neutron source was present.

TEST METHODOLOGY

Prior to initial installation of the equipment, the temporary detectors were positioned near a neutron source. Using the neutron source, an optimum operating voltage was selected for each of the three detectors to ensure that minor fluctuations in detector power supply voltages would not adversely affect detector output. With the individual detector operating voltages selected, discriminator voltages were determined based on detector characteristic curves. Following the initial installation of the temporary detectors, a submersible Am-Be neutron source was placed into the reactor vessel next to each detector to verify response to neutrons.

SUMMARY OF RESULTS

Upon completion of the procedure, operating voltages were determined to be 2100 volts for all three temporary nuclear instrumentation channels and discriminator voltages were set at 1.5 volts for all three temporary nuclear instrumentation channels. Three spare detectors were installed and the test was repeated with the same results. Each of the three temporary instrumentation channels responded by a positive change in indicated count rate.

3.1.2.b - CORE LOADING INSTRUMENTATION (NEUTRON CHECK DURING CORE LOADING), NR-30B

OBJECTIVE

The core loading instrumentation test procedure was performed to verify that the temporary instruments provided indication when a neutron source was present prior to resuming core loading following an eight hour delay in loading.

TEST METHODOLOGY

Following the suspension of core loading for eight hours, either a submersible neutron source was lowered into the vessel and moved toward each of the three temporary detectors, or a primary source bearing assembly was lifted from then lowered into the core. As the source approached each detector or an assembly was lifted the indicated count rate was observed.

SUMMARY OF RESULTS

The test procedure was executed in its entirety five times during the core loading activity. During each performance it was observed that as the source approached each detector or an assembly was moved, the count rate increased, indicating that the detector was responding.

During core load temporary detector failure occured five times. The failures were attributed to detector can seal leakage. This had no impact on core loading (other than schedule delays) since two detectors remained operable at all times and could be repositioned to optimize response.

3.1.3 - OPERATIONAL ALIGNMENT OF EXCORE NUCLEAR INSTRUMENTATION (PRIOR TO CORE LOAD, DURING CORE LOAD), NR-34A/B

OBJECTIVE

This test was performed to verify that the Source Range (SR) and Intermediate Range (IR) excore instrumentation channels were functioning as designed and capable of detecting and alarming upon an excessive insertion of positive reactivity. Test NR-34B was specifically used to re-verify the response of SR channels to a neutron source when core load procedures experienced a delay of eight hours or more.

TEST METHODOLOGY

Operational verification of the high level trips, high flux indications and containment evacuation horns was assured by observing annunciator windows on the main control board and SR/IR channel drawer. Each SR channel responded by indicating a positive change in the count rate when a neutron source was introduced near the detector associated with the channel. SR channel circuit components were adjusted.

SUMMARY OF RESULTS

The level trip bistables for SR channels N31 and N32 tripped at 10^5 cps and 1.1×10^5 cps, respectively. The value for N32 was greater than the expected value of $\leq 10^5$ cps but within the Technical Specification allowable value of 1.4×10^5 cps. The high flux at shutdown bistables for these channels tripped at 50 cps and 51 cps, respectively. Proper actuation of main control board annunciators, level trip and high flux at shutdown drawer windows, and containment evacuation horn was observed.

The level trip bistables for IR channels N35 and N36 tripped at 7.5×10^{-5} amps and 7.2×10^{-5} amps, respectively, and the power above P-6 bistable for both channels tripped at 1.1×10^{-10} amps. Proper actuation of power above permissive P-6 and high level trip drawer windows was observed. These numeric values met the acceptance criteria.

Testing conducted prior to core load utilized a 5.0 curie Am-Be neutron source. An increase in neutron response was observed when the neutron source was placed inside the reactor vessel near the detector. Subsequent detector response verification utilized a different neutron source methodology. Specifically, the vertical movement of a fuel assembly lying between the primary Californium source and the SR detector acted as a neutron source by subcritical multiplication of the Californium source neutrons. This methodology was utilized on four occasions. Each time a decrease in countrate was observed as the fuel assembly was raised from its position thereby removing its subcritical multiplication effect on the source neutrons.

3.1.4 - REACTOR SYSTEMS CHEMICAL SAMPLING FOR CORE LOAD, PS-30

OBJECTIVE

This test was performed to verify correct and uniform boron concentrations in unisolated portions of the reactor coolant system (RCS) and the directly connected portions of the auxiliary systems as required for core loading. This test was also designed to ensure that the possibility of an inadvertent dilution of the RCS during core loading was minimized.

TEST METHOLODOGY

Prior to the commencement of core loading, the RCS was sampled and verified to meet the specified water chemistry criteria stated in the test. This procedure was then repeated daily throughout the core loading process.

Each of the RCS cold legs, the safety injection system accumulators, the safety injection system, the containment spray system, and the spent fuel pool were sampled, if they contained water, and that water was verified to contain at least 2000 ppm boron. An exception to this was the U-1 boric acid storage tank which had a limit of > 7000 ppm boron.

Following the initial verification of the chemistry in the reactor coolant system, four samples were taken at equidistant depths from the reactor vessel along with a sample from the operating residual heat removal train. These samples were then analyzed for boron to verify a uniform boron concentration throughout the entire system. All samples were verified to be within 30 ppm of each other. Within four hours after the RCS boron was verified to be at a uniform concentration, the refueling canal inside and outside of containment and the operating residual heat removal (RHR) train were sampled and analyzed for boron to verify that all water within these systems were > 2000 ppm boron. Sampling continued on the refueling canal and the RHR train every four hours throughout the core loading process.

SUMMARY OF RESULTS

During the execution of this test which lasted throughout the core loading process, all acceptance criteria were adequately met for each system that was sampled. No corrective actions in the core loading process were needed to meet the acceptance of this test. All accumulators remained drained throughout the the core loading process along with the U-2 boric acid storage tank and the spent fuel pool. All other systems that were sampled were shown to contain a boron concentration of at least 2000 ppm. The U-1 boric acid storage tank had a concentration of 7122 ppm boron. These results assured that there was no inadvertant dilution of the RCS boron level during core load.

3.1.4 - REACTOR SYSTEMS CHEMICAL SAMPLING FOR CORE LOAD, PS-30 (Continued)

At no time during the execution of this test did the water in the RCS exceed the limits of 0.150 ppm chloride and fluoride. The lowest boron concentration found within that system during core load was 2070 ppm on 10/23/86. A maximum value of 2115 ppm was reached on 11/2/86.

The refueling canal inside and outside of containment had a boron concentration greater than 2000 ppm throughout the execution of this test.

3.1.5 - RADIATION SURVEYS PRIOR TO CORE LOADING, PS-31

OBJECTIVE

The radiation surveys prior to core loading test were performed to establish baseline data to be compared with radiation measurements during power ascension and commercial operation and to establish gamma and neutron radiation levels for the calibration facility.

TEST METHODOLOGY

Baseline radiation values were established by using precalibrated portable survey instrumentation to measure dose rates at a series of specified points throughout Braidwood Station. Gamma readings were taken throughout the Service, Radwaste, Turbine, Auxiliary, and Fuel Handling Buildings and the Unit 1 Containment.

SUMMARY OF RESULTS

All of the specified points were surveyed for gamma radiation. All readings were 0.02 mr/hr or less. This value is more than one order of magnitude less than the acceptance criteria limit from FSAR Table 12.3-1. None of the radiation base point measurements required further investigation.

All radiation readings were within the specified acceptance criteria with the exception of the Model 149 Neutron Calibration Unit. This was the result of a larger source being supplied than expected. However, the calibration unit met manufacturers limits and those specified in 10 CFR 20.204A. No other radiation base point measurements required further investigation.

3.2.1 - PIPE VIBRATION (PRE CRITICAL), EM-30A

OBJECTIVE

The Pipe Vibration test procedure demonstrated that the peak stresses resulting from steady state flow induced vibration were within allowable design limits. The scope of the test was limited to portions of the main steam system.

TEST METHODOLOGY

The system involved was operated under normal, steady state design conditions during which a visual inspection of the piping was conducted. The walkdown divided the system into smaller piping subsystems between restraints in order to utilize simple beam analogy to determine deflection limits. Portable vibration analyzers were used to obtain vibration levels and a comparison was made between the calculated deflection limit and the actual reading. Based on this comparison, vibration levels less than the allowable limit were deemed acceptable. If any levels had exceeded the allowable limit further analysis by offsite engineering would have been requested.

SUMMARY OF RESULTS

All piping was visually inspected. Those subsystems displaying the greatest vibration were subjected to further analysis. The final results of the analysis indicated that the subsystems met acceptance criteria in that the levels of vibration did not exceed the allowable design limits. No further offsite analysis was required.

3.2.2 - THERMAL EXPANSION - FEEDWATER, FW-32A

OBJECTIVE

Thermal expansion testing of the feedwater system was conducted to verify that components and piping could expand without restriction of movement upon system heatup. It was also conducted to confirm the correct functioning of component supports, piping supports and restraints.

TEST METHODOLOGY

At feedwater system ambient and hot conditions, system walkdowns were performed. Piping and components were visually examined and snubber positions recorded. Interferences were identified and dispositioned by the design engineers. When necessary, system walkdowns were again conducted following the resolution of interferences. All piping movements were evaluated by the design engineers.

SUMMARY OF RESULTS

The piping and components were not constrained from expanding and actual thermal expansion movements were verified to be within ± 25% or ± 1/4 inch of expected movements. During the course of system walkdowns, several minor interferences were identified. These interferences were evaluated by design engineers and determined to be acceptable as is, or specific corrective action was recommended. All recommended corrective actions were performed. Some portions of the feedwater system were again examined and measured following the removal of interferences. Movements of components not within the ± 25% or ± 1/4 inch criterion were evaluated by the design engineers on a case-by-case basis. All thermal expansion movements were found to be acceptable.

3.2.3 - WATER HAMMER PREVENTION, FW-33A

OBJECTIVE

The upper nozzle of the steam generator was tested for damaging water hammer following:

1) The initiation of auxiliary feedwater through an uncovered upper nozzle, and

2) Re-initiation of the purge flow through an uncovered upper nozzle.

TEST METHODOLOGY

With the reactor coolant system at no load temperature and pressure, the steam generator level was decreased to 55% to uncover the upper nozzle. After 30 minutes, a purge flow through the upper nozzle was established. A system test engineer was positioned to observe the upper nozzle water hammer. After that transient was observed, steam generator level was lowered again to 55%. Auxiliary feedwater was initiated with the engineer listening for any water hammer.

SUMMARY OF RESULTS

No damaging water hammer was observed during this test by the system test engineer.

3.2.4 - INCORE FLUX MAPPING SYSTEM SYSTEM CHECKOUT, IC-30

OBJECTIVE

The purpose of this procedure was to demonstrate the proper operation of the flux mapping system which includes the leak detection system and the CO_2 gas purge system. In addition, top and bottom of core limits were set and the actual drive cables and detectors were installed.

TEST METHODOLOGY

Using dummy drive cables, the top and bottom of core limits were established for normal, emergency, calibrate and storage modes by slowly driving the dummy detectors to the top of the core (or storage position) where clutch slippage was observed. The position was then recorded from the encoder display. The top limit was obtained by subtracting two inches from the recorded position and the bottom limit was obtained by subtracting 170 inches from the top limit. The leak detection system was tested by filling the drain header with demineralized water and allowing the leak detection level switch to actuate, thereby draining the water. The withdraw limit switch was bypassed to verify the safety limit switch would prevent the detector from being taken up onto the reel. All push-to-test lights were verified. The CO_2 gas purge system was checked to verify a positive pressure existed in the system when the detector cables were withdrawn.

SUMMARY OF RESULTS

Proper operation of all indicating lights were verified along with the proper operation of the leak detection system as described in the previous section. The CO_2 gas purge system was verified to produce a positive pressure in the system when the detector cables were withdrawn. The dummy detectors were inserted into all core locations. All top and bottom limits were established. The safety limit switches were demonstrated operable as previously discussed. As a final step, the actual detector cables were installed on the drive units.

3.2.5 - THERMAL POWER MEASUREMENT AND STATEPOINT DATA COLLECTION (PRE-CRITICAL) IT-32A

OBJECTIVE

The objective of this test was to check WT instrumentation and check Tave alignment during precritical testing. There was no acceptance criteria for this test,

TEST METHODOLOGY

Alignment check of the delta T and Tave process instrumentation was accomplished by checking values gathered at normal operating temperature and pressure conditions and comparing them to expected values.

SUMMARY OF RESULTS

The test was conducted at pre-critical conditions and identified minor out-of-tolerance conditions on the various delta T summing amplifiers. These conditions were a result of the Reactor Coolant Pump heat making the Tcold legs hotter than the Thot legs. This is an acceptable condition.

3.2.6 - INCORE THERMOCOUPLE/RTD CROSS CALIBRATION, IT-33

OBJECTIVE

The objective of the test was to verify reactor coolant system RTD performance over a range of temperatures. Additionally a cross calibration data base was developed for the incore thermocouple system and reactor vessel level indication system.

TEST METHODOLOGY

Data was gathered at four temperature plateaus; 250°, 340°, 450° and 557°F. The plant conditions required at each plateau included RCS temperature stabilization allowing for no more than ±0.5°F drift, steam generator levels within 10% of one another, all four reactor coolant pumps running and the plant in mode 3, 4 or 5. For the RTDs, four sets of data were taken at each plateau by monitoring RTD resistance at the master test cards with a four wire resistance meter. For the thermocouples, two sets of data were taken at each plateau from indicators located in the main control room and process computer trending logs. At each plateau a correction factor was calculated based on the average of the narrow range RTDs.

SUMMARY OF RESULTS

The initial calculations of the difference between RCS stabilized temperature and each RTD's indicated temperature were performed with preliminary calibration curves derived by Westinghouse from manufacturer's data. Upon applying these calibration curves to the test data, several narrow range RTDs were outside acceptable limits. Consequently, final calibration curves were generated using Westinghouse methodology and the test data.

Based on these final curves, all narrow range RTDs were recalibrated and found to be well within acceptable limits. The wide range RTDs and thermocouples were found to provide acceptable temperature indication.

3.2.7 - DIGITAL ROD POSITION INDICATION SYSTEM CHECKOUT, PI-30

OBJECTIVES

The digital rod position indication system checkout test procedure was performed to verify the following indication and alarm functions for each individual RCCA: system accuracy (full and half), rod position indication versus demand deviation alarm, rod at bottom indication, and data transmission. In the process of demonstrating the indication and alarms above, it was shown that all RCCAs operated over their full range of travel.

TEST METHODOLOGY

Two methods of testing were used to fully prove the rod position indication system in this test procedure. The first type of testing involved injecting simulated rod position data from the test switches in the data cabinets and observing the resulting LED indication locally and in the control room. The second type of testing involved using actual rod position data by moving individual RCCAs and observing indications and alarms that resulted. Individual rods were disconnected from their group using the manual lift disconnect switches in the main control room. All RCCAs were tested in this manner.

SUMMARY OF RESULTS

Accuracy requirements for the rod position indication system from the vendor technical manual were $\pm$ 4 steps in comparison with associated step counters. All control banks met this criterion over the entire range of rod travel from rod bottom (0 steps) to 228 steps. Shutdown banks met this criterion in the ranges from rod bottom to 18 steps and from 210 to 228 steps. The transition region of each shutdown bank was shown to indicate correctly between 21 $\pm$ 4 steps and 207 $\pm$ 4 steps as indicated on the step counters. Rod bottom indication was shown to occur for all shutdown and control rods at 3 $\pm$ 1 steps on the step counters. The Rod Versus Bank Deviation, Shutdown Rod Off Top, and Rod At Bottom annunciator alarms were demonstrated to operate correctly, as well.

3.2.8 - CHEMISTRY AND RADIOCHEMISTRY FOR MONITORING WATER QUALITY DURING STARTUP AND POWER ASCENSION, PS-32

OBJECTIVE

This test was performed to verify that the water quality within the primary water make-up system and the reactor coolant system met the chemistry requirements specified in the Technical Specifications and/or the Westinghouse NSSS guidelines prior to criticality.

TEST METHODOLOGY

The testing was performed by obtaining samples of the primary water make-up and reactor coolant systems from the appropriate sample panels throughout the plant. Chemical analyses were then performed on the sample from each system. The results of these analyses were tabulated, and compared to acceptance criteria values.

SUMMARY OF RESULTS

All acceptance criteria were adequately met for each system that was sampled. No corrective actions in plant operation were needed to meet the acceptance criteria.

Tables 3.2.8-1 and 3.2.8-2 contain a summary of the results for each system sampled along with the acceptance criteria or guidelines stated within the test.

TABLE 3.2.8-1

SUMMARY OF REACTOR COOLANT CHEMISTRY DURING EACH EXECUTION OF STARTUP TEST PS-32

CHEMISTRY PARAMETER	CRITERIA/ GUIDELINE	MODE 3
Chloride	<150 ppb	<5
Fluoride	<150 ppb	<10
Dissolved* Oxygen	<100 ppb	9.2
Silica	<1000 ppb	613
Lithium	0.7-2.2 ppm	1.24
Hydrogen**	25-50 cc/kg	N/A
Suspended Solids	<1.0 ppm	<0.2
Aluminum	<50 ppb	7.8
Calcium	<50 ppb	2.9
Magnesium	<50 ppb	1.9

* When Tave > 180°F

** When RCS > 1% Reactor Power

TABLE 3.2.8-2

SUMMARY OF PRIMARY WATER MAKEUP CHEMISTRY DURING EACH EXECUTION OF STARTUP TEST PS-32

CHEMISTRY PARAMETER	CRITERIA/ GUIDELINE	MODE 3
Total Chloride and Flouride	<100 ppb	5.6
Silica	<100 ppb	9.3
pH @ 25 C	6.0-8.0	6.06
Specific Conductivity Micromho/cm @ 25°C	<1.0	0.4714
Suspended Solids	<0.1 ppm	<0.05
Aluminum	<20 ppb	<1.0
Calcium	<20 ppb	<1.0
Magnesium	<20 ppb	<2.0
Potassium	<10 ppb	<2.0

3.2.9 - RTD BYPASS LOOP FLOW VERIFICATION RC-30

OBJECTIVE

The RTD Bypass Loop Flow Verificationtest procedure was performed to verify the actual hot leg RTD bypass loop flowrates were greater than the flowrate required to meet a 1.0 second transit time, which is used in the design assumptions.

TEST METHODOLOGY

Required minimum flow for a one second transit time was first calculated from actual pipe dimensions. Next, actual flowrates were determined for all four hot & cold leg RTD's by isolating the hot leg and recording the cold leg value and then repeating the steps for determining the hot leg flow. These measured values were then compared to the minimum valves to verify acceptability.

SUMMARY OF RESULTS

The calculated flow rates derived from the measured values were less than minimum required flow for the hot leg RTD's in all cases. A Safety Evaluation was performed for the results of this test. Factors other than fluid transport time contribute to the overall RTD response time. This overall response time will be measured during the plant trip from 100% power. AIR number 456-509-87-R20-87-052 will ensure the verification of total loop response time and the acceptability of the data collected per this procedure.

HOT LEGS

Loop No	Calculated Flow 1 (gpm)	Required Flow 2 (gpm)	Difference Cal.-Req. (Greater than 0)	Actual Transit Time (sec)
1	111.4	123	11.6	1.10
2	111.4	123	11.6	1.10
3	121.1	127	5.6	1.05
4	108.5	123	14.5	1.13

1. Based on total, cold & hot leg measured flows.

2. Based on actual pipe lengths and diameters to give a one second transit time.

3.2.10 - REACTOR COOLANT SYSTEM (RCS) FLOW MEASUREMENT (HOT STANDBY), RC-31A

OBJECTIVE

The RCS Flow Measurement at Hot Standby test procedure was performed to determine the RCS flowrates for each of the 4 loops and then the total flowrate. Also the reactor coolant pumps (RCPs) vibration measurements were taken from the installed vibration pickups during performance of the test.

TEST METHODOLOGY

Prior to criticality data was obtained from the installed elbow tap differential pressure (d/p) instrumentation and used to find the RCS flowrates. Data was recorded every minute for ten minutes which included the following: RCS cold leg RTD resistance readings, and d/p transmitter output voltage. Then the RTD resistance readings were converted to temperature (°F) and the d/p transmitter output voltage readings were converted to in. H_2O at 68°F. Each loop had three flow transmitters from which a d/p measurement was taken. The d/p readings (in. H_2O at 68°F) were used to determine three flowrates for each loop by using the Elbow Tap vs Reactor Coolant Cold Leg Volumetric Flow Rate figure. These three flowrates were averaged to obtain the loop average flow, and then all four loops average flowrates were added to obtain the total RCS flowrate.

SUMMARY OF RESULTS

The total RCS flowrate must be equal to or greater than 339,840 gpm, which is 90% of the thermal design flow, as determined by elbow tap d/p prior to criticality. The expected flowrate for each loop was greater than or equal to 84,960 gpm with the actual measured average loop flowrates being: Loop 1 = 104,400 gpm; Loop 2 = 102,133 gpm; Loop 3 = 101,700 gpm; and Loop 4 = 101,367 gpm. The sum of the average loop flowrates gives the total RCS flowrate of 418,032 gpm.

3.2.11 - RC FLOW COASTDOWN, RC-32

OBJECTIVE

The RC Flow Coastdown test was performed with the unit at Hot Standby to verify the measured core flow exceeds or is equal to the flow assumed in the accident analysis. In addition, time delays for low flow, undervoltage and underfrequency trips were verified to be within acceptable limits.

TEST METHODOLOGY

Strip chart recorders were connected to the solid state protection system to monitor reactor coolant flow characteristics and reactor trip breaker positions. Data from the strip charts were then plotted on various graphs to verify acceptability of the measured flow values and time delays.

SUMMARY OF RESULTS

Time delays for the underfrequency and undervoltage trips met the required maximum values of 0.6 and 1.5 seconds, respectively. Actual values obtained were 0.431 seconds for the underfrequency trip and 1.0405 seconds for the undervoltage trip.

The time delay for the low flow trip was calculated to be 1.08 seconds. This did not meet the acceptance criteria of 1.0 seconds.

A review of the calculations resulted in a more accurate determination of the gripper release time measurement. Based upon the revised methodology, time delays were recalculated to be 0.87, 0.216, and 0.826 seconds for the low flow, underfrequency, and underfrequency trips, respectively. These values meet the acceptance criteria values stated above.

In addition, the reactor coolant pumps were verified to trip within 100 msec of each other. Actual value obtained was 2 msec. This was acceptable.

The measured flow coastdown time constant, 12.99, was found to be greater than the minimum design value of 11.87.

3.2.12 - REACTOR COOLANT SYSTEM LEAK TEST, RC-33

OBJECTIVE

The purpose of this procedure was to verify the reactor coolant system (RCS) leak tightness after the system had been closed.

TEST METHODOLOGY

With the plant in in Hot Standby conditions, prior to initial criticality, the reactor coolant system was tested to verify leak tightness. The reactor coolant pressure was increased to between 2345 and 2355 psig for testing purposes. This was accomplished by adjusting the Master Pressurizer Pressure Controller and energizing the Pressurizer Heaters. After the pressure had stabilized, the leak test was conducted by thorough visual inspection. In this inspection, the reactor pressure vessel, pressurizer and all four reactor coolant loops were verified for leak tightness. In addition, the reactor cavity sump, reactor coolant drain tank and containment floor sump levels were measured. Also, the unidentified, identified, and controlled leakage rates were determined.

SUMMARY OF RESULTS

The acceptance criteria for this test was to verify that RCS leakage would be: less than or equal to 1 gpm unidentified leakage, less than or equal to 10 gpm identified leakage, less than or equal to 40 gpm controlled leakage, and no pressure boundary leakage. Through the visual inspection, no pressure boundary leakage was observed. Included in this were observations of the containment floor sump and the reactor cavity sump. No leakage was indicated (ie, no increase in levels), and during the test duration, neither of the containment floor sump pumps nor the reactor cavity sump pump were run. The unidentified leakage was determined to be 0.6828 gpm. The identified leakage was found to be 0.00087 gpm. Finally, the controlled leakage was measured for each reactor coolant pump seal injection flow and totalled to 37.5 gpm. This test thus verified acceptable leak tightness of the reactor coolant system.

3.2.13 - CONTROL ROD DRIVE MECHANISM OPERATION TEST, RD-30

OBJECTIVE

The test objectives were to verify the proper slave cycler timing, to perform an operational check of each control rod drive mechanism (CRDM) with a rod cluster control assembly (RCCA) attached prior to initial use of the mechanism under the conditions stated below and to demonstrate that the auto and manual rod withdrawal blocks functioned as designed.

TEST METHODOLOGY

The test was performed under two plant conditions: Mode 5 - cold, with one RCP running and Mode 3 - hot, full flow. The proper operation of the slave cycler timing was verified under cold conditions with one RCP running. This was a change from the test requirements of no flow. This was deemed to have no effect on the test results. However, a safety evaluation was performed in accordance with 10 CFR 50.59 and the NRC was notified of this change. The CRDM operational check was performed under both conditions, and the auto and manual rod withdrawal blocks were tested only under hot, full flow conditions.

Slave cycler timing was performed with all rods positioned at the core bottom. Initially, a single rod was withdrawn 50 steps to ensure the rodlet tips were at least 6 steps above the constricted thimble tube dashpot region. The rod was withdrawn 6 steps and then inserted 6 steps, while slave cycler timing data was recorded. The data monitored included lift coil currents, movable and stationary gripper currents, and a sound pickup from a carbon microphone attached to the top end of the rod travel housing. The tested rod was inserted to the bottom of the core and the above procedure was repeated for each power cabinet.

The CRDM operational check was performed with all rods initially at the core bottom. A single rod was then withdrawn 48 steps. Next, data was recorded while the rod was withdrawn 12 steps, and then inserted 12 steps. The same parameters were monitored here as in the slave cycler timing procedure. This was repeated for both control and shutdown rods.

Auto and manual rod withdrawal blocks were also tested. A signal simulating 50% nuclear power was fed to the power range channels, and the appropriate signals for turbine first stage pressure and Tave generated. By generating a T error signal and using jumpers and tripping various bistables, the following rod blocks & alarms were checked:

3.2.13 - CONTROL ROD DRIVE MECHANISM OPERATION TEST, RD-30 (Continued)

1) Low Power Block of Automatic Rod Withdrawal (C-5)
2) Auto Rod Withdrawal Block When Control Bank D is Above Withdrawal Limit (C-11)
3) Intermediate Range High Flux Rod Stop (C-1)
4) Power Range High Flux Rod Stop (C-2)
5) OTWT Rod Stop & Turbine Runback (C-3)
6) OPWT Rod Stop & Turbine Runback (C-4)
7) Low & Low Low Insertion Limits for Control Banks A, B, C and D

SUMMARY OF RESULTS

The slave cycler timing traces were found to be acceptable for all five power cabinets. The operability of the CRDM was verified for all 53 CRDMs by demonstration of control rod insertion and withdrawal.

3.2.14 - ROD CONTROL SYSTEM CHECKOUT, RD-31

OBJECTIVE

The rod control system checkout test demonstrated that the rod control system performed the required control and indication functions in the areas listed below and verified it was ready for use just prior to initial criticality.

1. Individual Bank Rod Motion Test
2. Bank Overlap Operation Checkout
3. Non-Urgent and Urgent Alarms for Power Cabinets and Logic Cabinets
4. D.C. Hold Cabinet

TEST METHODOLOGY

Each rod control cluster (RCC) bank was singly withdrawn from the core 48 steps, inserted to 8 steps, and then fully inserted to 0 steps. This was done to verify the proper operation of the demand step counter, digital rod position indication (DRPI) system and proper annunciator response. Bank overlap was set to 12 steps and all shutdown banks were withdrawn to 30 steps. Control banks were withdrawn until control bank D (CBD) was at 20 steps to demonstrate proper overlap. All RCC banks were then reinserted. The non-urgent alarm was tested for each cabinet by pulling fuses on one of the two redundant power supplies. The urgent alarm was tested for each cabinet by pulling a circuit card. The D.C. hold cabinet was checked to verify it was capable of supplying the necessary current to the stationary gripper coils.

SUMMARY OF RESULTS

The rod motion indicator (ROD IN/ROD OUT) operated satisfactorily for all nine RCC banks. The demand step counters functioned properly for all RCC banks. Rod speed indication was acceptable for each RCC bank and all speeds were acceptable. Bank overlap performed properly. Non-urgent and urgent annunciation in the control room was completed successfully. The proper current reduction was shown upon an urgent failure. The D.C. hold cabinet exceeded its acceptance criteria by supplying too high a voltage to the stationary gripper coil. A station action item was generated, and then reviewed by offsite engineering. The test results were accepted based on Westinghouse evaluation.

3.2.15 - ROD DROP TIME MEASUREMENT, RD-33

OBJECTIVE

The purpose of this procedure was to determine the amount of time required to drop each RCCA from its fully withdrawn position 228 steps, to the point of entry into the dashpot region. This was done in accordance with Technical Specification 3.1.3.4 which states the maximum drop time as 2.4 seconds for hot full flow condition.

TEST METHODOLOGY

With all rods fully inserted and the RCS Boron Concentration greater than 2000 ppm, one Bank of Rods was withdrawn to 288 steps. Using the Auto Rod Drop System, the Digital Rod Position indication (DRPI) was deenergized and rods were dropped in their respective group. The Data "A" and Data "B" coil signals were collected by the Auto Rod Drop System Remote units and the data transferred to the cart. The signals were added and plotted by the cart on a graphics capable thermal printer. All traces were verified to be less than 2.4 seconds. Any rod drop time that was outside of two standard deviations from the average were dropped three additional times for each rod. Proper operation of the dashpot was also verified from the traces. This test was performed with the plant at hot full flow condition.

SUMMARY OF RESULTS

All rods in the hot full flow test condition dropped in less than 2.4 seconds. The following times were recorded for the hot full flow condition.

	Average	Fastest	Slowest
Hot Full Flow	1.463 Sec.	1.412 Sec.	1.508 Sec.

At test conditions, the voltage traces were examined and proper operation of the dashpot region was verified.

3.2.16 - REACTOR PROTECTION LOGIC, RP-30

OBJECTIVE

The objective of the test was to verify proper operation of the automatic and manual reactor trip circuitry and to verify proper operation of the reactor trip breakers. The test was also designed to verify proper operation of the permissive and block circuitry.

TEST METHODOLOGY

The test verified on both trains, with the use of the semi-automatic logic tester, proper operation of the reactor trip system logic combinations employed in the solid state protection system (SSPS). The test also verified proper operation of the reactor trip breakers in response to automatic and manual reactor trip signals. Additionally the test verified proper operation of the P-4 permissive and the resistance from logic ground to chassis ground in each train of the SSPS.

SUMMARY OF RESULTS

The test proved SSPS logic combinations for reactor trip inputs as well as permissive and block inputs. Also, the test showed that the reactor trip breakers were operational when a signal was received from either an automatic or manual trip signal. Finally the test verified proper operation of the P-4 permissive and the SSPS ground circuits.

3.2.17 - PRESSURIZER SPRAYS, HEATERS AND BYPASS FLOW ADJUSTMENTS, RY-30

OBJECTIVE

The Pressurizer Sprays, Heaters and Bypass Flow Adjustments test was performed to verify pressurizer spray and heater effectiveness. In addition, spray line bypass valves were adjusted to maintain spray line temperature above 540°F.

TEST METHODOLOGY

In order to set the spray line flow, the valves were closed and the temperatures allowed to stabilize. The valves were then opened in 1/16 turn increments until a satisfactory temperature reading was achieved. To verify spray effectiveness, the heaters were manually isolated and the spray valves placed in the full open position. Pressurizer parameters were monitored via strip chart recorders. These parameters were then analyzed and plotted to verify the pressure transient fell within the allowable limits.

To verify heater effectiveness, the spray valves were manually isolated and the heaters were placed to the full on position. Pressure parameters were monitored via strip chart recorders. These parameters were then analyzed and plotted to verify the pressure transient fell within the allowable limits.

SUMMARY OF RESULTS

The spray valves were set to 1/16 turn and 5/16 turn to allow enough bypass flow to maintain spray line temperature above 540°F. The pressure transient resulting from the spray effectiveness test fell within the required band when adjustments were made for absolute pressure.

The pressure transient resulting from the heater effectiveness tests fell outside the required band for the first 40 seconds of the transient. The data was also adjusted for absolute pressure. The test results were evaluated by Westinghouse and PED and found acceptable.

3.2.18 - HEAT CAPACITY VERIFICATION FOR DIESEL GENERATOR VENTILATION VD-30

OBJECTIVE

The Heat Capacity Verification for Diesel Generator Ventilation procedure was performed to provide heat removal capacity data for the diesel generator ventilation system. The data obtained in the procedure was analyzed to verify that the design heat loads could be removed.

TEST METHODOLOGY

In order to obtain data for the heat removal capacity of the diesel generator ventilation system, each diesel generator room was tested independently with the associated diesel generator operating. The temperatures, heat loads, and cooling air temperatures for each diesel generator room were measured with the ventilation system in a recirculation mode. The data for each diesel generator room ventilation system was then analyzed to determine that the design heat loads could be removed.

SUMMARY OF RESULTS

Offsite engineering analyzed the data and determined that the systems would be able to remove the design heat loads. Since the diesel generators were operating under load and the ventilation fans were operating in 100% recirculation mode, the systems were operated near design conditions. The 1A diesel generator ventilation system maximum room temperature was 102°F with a supply air temperature of 94°F. The 1B diesel generator ventilation system maximum room temperature was 97°F with a supply air temperature of 90°F. This demonstrated the ability of the diesel generator ventilation system to limit the ambient room temperature to 130°F.

3.2.19 - OPERATIONAL ALIGNMENT OF EXCORE NUCLEAR INSTRUMENTATION (PRIOR TO INITIAL CRITICALITY) NR-34C

OBJECTIVE

This test was performed to verify that the excore nuclear instrumentation system was functioning per design and capable of detecting, alarming and mitigating unplanned reactivity excursions prior to initial criticality.

TEST METHODOLOGY

Selected parameters and alarms were evaluated, monitored, and determined.

The proper setpoints, alarms and trip functions were verified for the Source Range (SR), Intermediate Range (IR) and Power Range (PR) channels.

SUMMARY OF RESULTS

The setpoints of the SR channel trips, IR channel trips, and PR channel trips were verified as meeting all associated acceptance criteria.

3.2.20 - INITIAL CRITICALITY, NR-31

OBJECTIVE

This test procedure provided a method by which initial criticality could be obtained in a cautious and controlled manner. The sequence, frequency, and core conditions for collecting nuclear data was provided as well as the method of analysis of this data. In addition, a manual reactor trip was demonstrated prior to initial criticality in accordance with the test procedure.

TEST METHODOLOGY

Initial conditions were established with the RCS at an average temperature of 555.5°F, RCS pressure at 2238 psig, RCS boron concentration greater than 2000 ppm, and all RCC banks fully inserted.

Programs were initiated to monitor neutron flux, reactivity and various other plant parameters for the duration of the test.

Reference counts were determined for each source range channel by collecting 10 sets of reference count data over 100 second intervals. These values were to be used in the ICRR (Inverse Count Rate Ratio) calculations after reactivity additions were made.

A manual reactor trip from 5 steps withdrawn was required to be performed to verify the control rod drive mechanisms would unlatch upon opening of the trip breakers. Shutdown banks were withdrawn in alphabetical order. Control banks were then manually withdrawn in an overlap configuration. Rod withdrawal was completed when CBD was positioned at 150 steps. The withdrawals were made in increments of 50 steps or less and the value of the ICRR was determined prior to continuing.

The remaining reactivity insertion required to achieve criticality was made by diluting the RCS boron concentration by additions of primary water to the RCS. The dilution rate was initially 60 gpm until the ICRR value fell below 0.3 and was then reduced to 30 gpm after allowing time for RCS mixing. The ICRR was renormalized at this point as well. This dilution was maintained until the renormalized ICRR value fell below 0.3 at which time the RCS dilution was terminated to allow for mixing. Criticality was achieved during this time period. Flux level was established at 5×10^{-9} to 5×10^{-8} amps using CBD.

3.2.20 - INITIAL CRITICALITY, NR-31 (Continued)

SUMMARY OF RESULTS

Acceptance Criteria were met in that the reactor was successfully tripped, criticality was achieved within the approximate range of the predicted boron concentration, and neutron flux level was established within specified bounds on the Intermediate Range NIS channels.

4.0 - REFERENCES

1) Braidwood Station Final Safety Analysis Report

2) Regulatory Guide 1.68, Revision 2

3) Braidwood Station Technical Specifications

4) Braidwood Station Operating License

5) Byron Station Unit 1 Cycle 1 Startup Report

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