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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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February 28, 1995
GO2-95-042

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
ANNUAL OPERATING REPORT 1994

References: 1) Title 10, Code of Federal Regulations, Part 50.59 (b)

2) WNP-2 Technical Specifications, 6.9.1.4 and 6.9.1.5

3) Regulatory Guide 1.16, Reporting of Operation Information, Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Operating Report for calendar year 1994. Should you have any questions or desire additional information regarding this report, please contact Mr. D. A. Swank at (509) 377-4563.

Sincerely,

J. V. Parrish (Mail Drop 1023)
Vice-President, Nuclear Operations

JDA/MAN/ml
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WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1994

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

Washington Public Power Supply System
3000 George Washington Way
Richland, Washington 99352

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1.0 INTRODUCTION

The 1994 Annual Operating Report of Washington Public Power Supply System Plant Number 2 (WNP-2) is submitted pursuant to the requirements of Federal Regulations and Facility Operating License NPF-21. Plant WNP-2 is a 3323 MWt, BWR-5, which began operation on December 13, 1984.

Following a record 257 consecutive days of operation, the plant was manually scrammed on April 26, 1994 when Control Room Operators observed slight fluctuations in the power range monitors. During a followup assessment it was determined that the cause was due to a malfunctioning valve position transmitter in the Reactor Recirculation Flow Control System. The small changes in recirculation flow control valve position caused corresponding changes to reactor coolant flow and power parameters.

On April 30, 1994 the plant officially entered the R-9 1994 maintenance and refueling outage. Following the outage, the plant was restarted on July 30, 1994. In the remaining months of the year, the plant ran at, or near, 100 percent power.

During 1994 there were several examples of major accomplishments which required significant effort on the part of Supply System personnel to complete. The following is a summary of those efforts.

The ninth refueling outage was successfully completed. Significant planned and emergent activities included:

- Replacement of 156 fuel assemblies.
- Application of a mechanical stress improvement program to 45 welds on 25 reactor nozzles.
- Core shroud weld inspection.
- A complete overhaul of the main generator.
- Performance of the 10-year ASME Code hydrostatic test.
- Performance of integrated and bypass leak rate tests.
- Jet pump hold-down beam replacement.
- Replacement of 10 and re-work of 25 electrical penetration modules that contained evidence of corrosion.

- Jet pump sensing line crack evaluation.
- Local power range monitor change out.
- Control Rod replacement with low cobalt materials

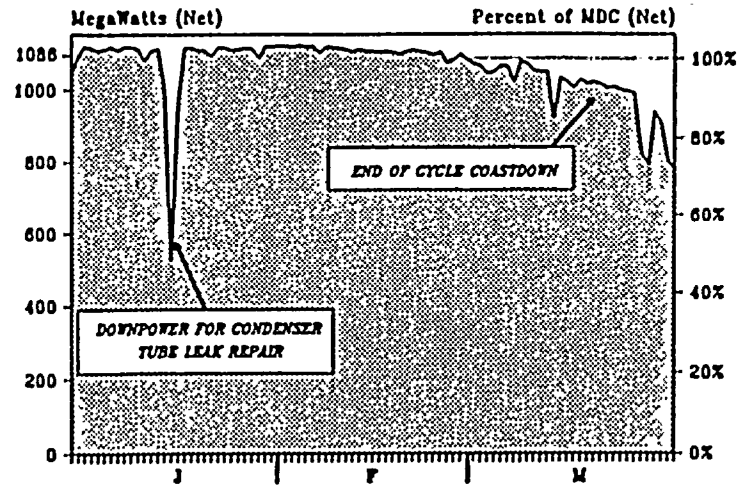
The best operating cycle of 257 days of continuous operation ended when the plant shutdown on April 26, 1994. During this cycle, calendar year 1994 was the second-best electrical generation year in the history of WNP-2. In 1994, gross generation was 7,041,290 megawatt-hours; net generation was 6,739,749 megawatt-hours; plant capacity factor was 70.8 percent; and plant availability factor was 74.2 percent.

On December 13, 1994, WNP-2 celebrated 10 years of commercial operation.

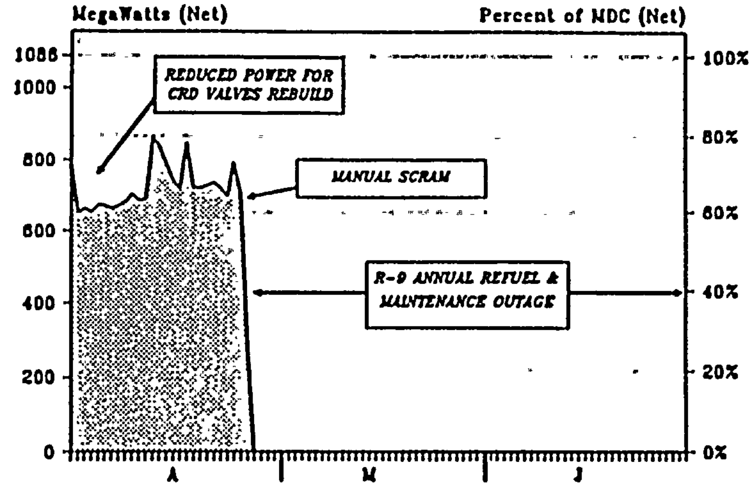
The 1994 capacity factors, based on net electrical energy output are listed below.

Month	Capacity Factor
January	99.76
February	101.76
March	91.66
April*	56.08
May	0
June	0
July**	0
August	97.94
September	100.98
October	101.59
November	101.97
December	102.29
Overall	70.85

WNP-2 LOAD PROFILE - CALENDAR YEAR 1994



1st QUARTER 1994



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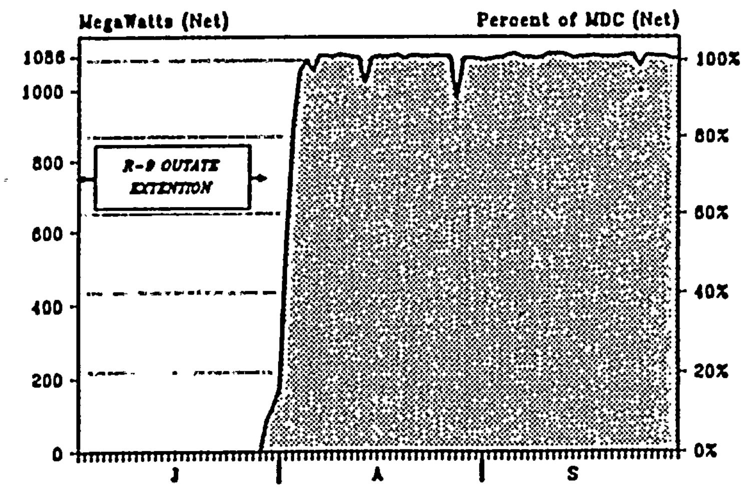

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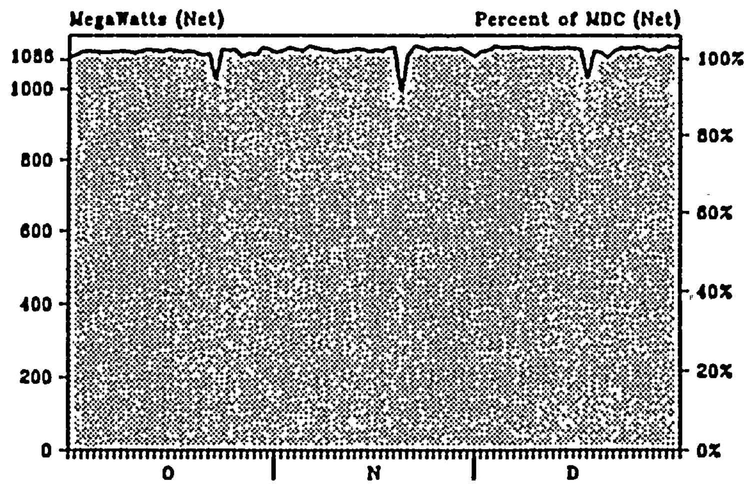


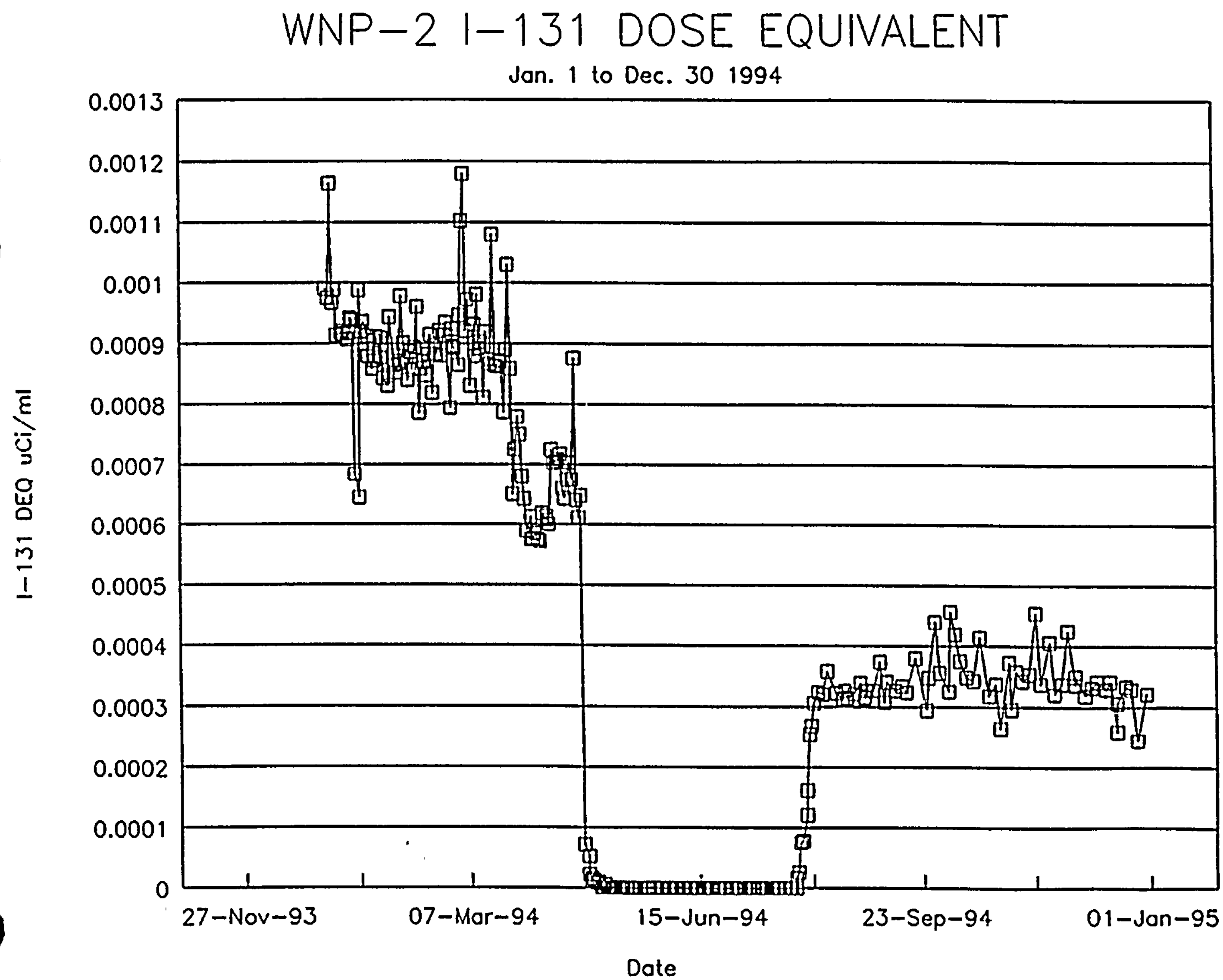
4th QUARTER 1994

1.1 WNP-2 LOAD PROFILE FOR 1994

1.2 REACTOR COOLANT SPECIFIC ACTIVITY LEVELS

This section contains information relative to reactor coolant cumulative iodine levels, iodine spikes and specific activity of all isotopes other than iodine, and is reported in accordance with Technical Specification paragraph 6.9.1.5.c.

The specific activity of the primary coolant was significantly less than 0.2 microcuries per gram dose equivalent I-131 as set forth in WNP-2 Technical Specification LCO 3/4.4.5. In addition, as shown below, the specific activity of the primary coolant was routinely sampled and was, in all cases, less than 100/E-bar microcuries per gram.



2.0 REPORTS

The reports provided in this section meet the requirements of Federal Regulations and the WNP-2 Operating License. They cover the requirements of the WNP-2 Technical Specifications, Sections 6.9.1.4 and 6.9.1.5 and provide the information specified by Regulatory Guide 1.16, "Reporting of Operating Information." In addition, Section 2.6 provides the information required by 10CFR50.59, "Changes, Tests, and Experiments."

2.1 ANNUAL PERSONNEL EXPOSURE AND MONITORING REPORT

The information provided in this section of the report is required by the WNP-2 Technical Specifications, Section 6.9.1.5a, and Regulatory Guide 1.16, Revision 4. These values are estimated doses for the listed activities based on pocket dosimetry readings. No correction has been applied to these readings

WASHINTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Facility: 02

Report for Calendar Year 1994

This report was produced with pocket dosimeter data

Number of persons receiving over 100 mrem is 1224. Total MAN-REM: 840.358

		Number of individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and others	Station Employees	Utility Employees	Contractor and others
OPERATIONS & SURVEILLANCE	Maintenance Personnel	46.60	2.32	10.31	27.625	1.337	5.158
	Operating Personnel	52.13	4.01	0.60	45.674	1.099	0.344
	Health Physics Personne	14.44	0.00	1.68	5.038	0.000	0.975
	Supervisory Personnel	19.26	3.67	2.31	4.994	0.818	0.324
	Engineering Personnel	13.26	23.18	7.18	2.662	6.562	2.282
ROUTINE MAINTENANCE	Maintenance Personnel	180.49	6.01	395.45	160.750	6.867	289.165
	Operating Personnel	3.42	0.02	0.40	5.542	0.161	0.185
	Health Physics Personne	31.78	0.00	64.69	17.788	0.000	34.417
	Supervisory Personnel	5.50	1.97	11.01	5.091	0.504	12.789
	Engineering Personnel	11.69	16.06	53.96	5.055	10.605	35.419
INSERVICE INSPECTION	Maintenance Personnel	9.91	0.57	7.86	4.771	0.336	5.179
	Operating Personnel	0.43	0.00	0.00	0.391	0.000	0.000
	Health Physics Personne	1.01	0.00	0.35	0.758	0.000	0.158
	Supervisory Personnel	0.59	0.06	1.07	0.149	0.002	0.369
	Engineering Personnel	1.95	2.30	8.99	0.564	1.021	2.509
SPECIAL MAINTENANCE (HSIP on RPV Nozzles)	Maintenance Personnel	18.37	0.28	48.27	18.390	0.207	32.665
	Operating Personnel	0.06	0.00	0.00	0.209	0.000	0.000
	Health Physics Personne	1.49	0.00	1.13	0.592	0.000	0.837
	Supervisory Personnel	1.11	0.59	0.06	0.684	0.025	0.064
	Engineering Personnel	0.93	3.83	4.09	0.332	1.995	2.525
SPECIAL MAINTENANCE (Replace Drywell Elec Penentrations)	Maintenance Personnel	2.41	0.04	6.34	2.415	0.027	4.290
	Operating Personnel	0.01	0.00	0.00	0.027	0.000	0.000
	Health Physics Personne	0.20	0.00	0.15	0.078	0.000	0.110
	Supervisory Personnel	0.15	0.08	0.01	0.090	0.003	0.008
	Engineering Personnel	0.12	0.50	0.54	0.044	0.262	0.332

WASHINTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Facility: 02

Report for Calendar Year 1994

This report was produced with pocket dosimeter data

Number of persons receiving over 100 mrem is 1224. Total MAN-REM: 840.358

		-----Number of individuals------			--------Year to Date Dose------		
		Station Employees	Utility Employees	Contractors and others	Station Employees	Utility Employees	Contractor and others
SPECIAL	Maintenance Personnel	1.86	0.03	4.89	1.863	0.021	3.309
MAINTENANCE	Operating Personnel	0.01	0.00	0.00	0.021	0.000	0.000
(Remove&Replace	Health Physics Personne	0.15	0.00	0.11	0.060	0.000	0.085
CEP/CSP Valves)	Supervisory Personnel	0.11	0.06	0.01	0.069	0.002	0.006
	Engineering Personnel	0.09	0.39	0.41	0.034	0.202	0.256
SPECIAL	Maintenance Personnel	0.33	0.00	0.87	0.331	0.004	0.588
MAINTENANCE	Operating Personnel	0.00	0.00	0.00	0.004	0.000	0.000
(Replace Jet Pump	Health Physics Personne	0.03	0.00	0.02	0.011	0.000	0.015
Beam Bolts)	Supervisory Personnel	0.02	0.01	0.00	0.012	0.000	0.001
	Engineering Personnel	0.02	0.07	0.07	0.006	0.036	0.045
WASTE	Maintenance Personnel	13.00	5.71	0.81	4.087	1.416	0.193
PROCESSING	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personne	0.95	0.00	5.36	0.367	0.000	3.079
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.02	0.17	0.08	0.001	0.015	0.030
REFUELING	Maintenance Personnel	24.26	0.02	12.71	26.552	0.014	4.697
	Operating Personnel	5.32	0.97	0.00	4.439	0.091	0.000
	Health Physics Personne	2.52	0.00	14.25	1.343	0.000	9.437
	Supervisory Personnel	1.73	0.53	0.07	1.130	0.062	0.001
	Engineering Personnel	2.76	4.66	7.92	0.761	1.288	1.836
TOTAL	Maintenance Personnel	297.23	14.98	487.51	246.784	10.229	345.245
	Operating Personnel	61.37	5.00	1.00	56.307	1.351	0.529
	Health Physics Personne	52.56	0.00	87.75	26.035	0.000	49.113
	Supervisory Personnel	28.47	6.97	14.53	12.220	1.417	13.563
	Engineering Personnel	30.84	51.17	83.25	9.458	21.986	45.233
Grand Total		470.47	78.12	674.04	350.804	34.983	453.683

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information pertaining to main steam line safety/relief valve challenges for calendar year 1994 in accordance with the requirements of WNP-2 Technical Specification 6.9.1.5(b).

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0136	63790-00-0062	63790-00-0135	63790-00-0137	63790-00-0055
Component ID (Location)	MS-RV-5B	MS-RV-5C	MS-RV-4A	MS-RV-4B	MS-RV-4C
Date of Actuation (Mo/Da/Yr)	7/27/94	7/27/94	7/28/94	7/28/94	7/28/94
Time of Day (24 Hour Clock)	1638	1638	1253	1254	1258
Type of Actuation (Code)	C	C	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	≈3	≈3	≈15	≈15	≈15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	920	920	931	931	931
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	4 Sec.	4 Sec.	29 Sec.	20 Sec.	23 Sec.
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached?					

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES (CONTINUED)

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0122	63790-00-0134	63790-00-0138	63790-00-0137	63790-00-0135
Component ID (Location)	MS-RV-2C	MS-RV-2B	MS-RV-2D	MS-RV-4B	MS-RV-4A
Date of Actuation (Mo/Da/Yr)	7/27/94	7/27/94	7/27/94	7/27/94	7/27/94
Time of Day (24 Hour Clock)	1638	1638	1638	1638	1638
Type of Actuation (Code)	C	C	C	C	C
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	≈3	≈3	≈3	≈3	≈3
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	920	920	920	920	921
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	8 Sec.	4 Sec.	4 Sec.	4 Sec.	4 Sec.
Failures, Reports (Code)	B	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached?					

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES (CONTINUED)

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0051	63790-00-0122	63790-00-0053	63790-00-0136	63790-00-0126
Component ID (Location)	MS-RV-2A	MS-RV-2C	MS-RV-3B	MS-RV-5B	MS-RV-3D
Date of Actuation (Mo/Da/Yr)	7/28/94	7/28/94	7/28/94	7/28/94	7/28/94
Time of Day (24 Hour Clock)	1319	1320	1321	1353	1401
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	≈15	≈15	≈15	≈15	≈15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	931	931	931	931	931
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	28 Sec.	14 Sec.	16 Sec.	55 Sec.	57 Sec.
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached?					

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES (CONTINUED)

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0120	63790-00-0047	63790-00-0052	63790-00-0138	63790-00-0057
Component ID (Location)	MS-RV-1B	MS-RV-1D	MS-RV-3C	MS-RV-2D	MS-RV-3A
Date of Actuation (Mo/Da/Yr)	7/28/94	7/28/94	7/28/94	7/28/94	7/28/94
Time of Day (24 Hour Clock)	1455	1538	1542	1545	1556
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	
Rx Power Level Prior to Lift (% Rated Thermal)	≈15	≈15	≈15	≈15	≈15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	931	931	931	931	931
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	7 Sec.	13 Sec.	11 Sec.	18 Sec.	46 Sec.
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached?					

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES (CONTINUED)

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0062	63790-00-0060	63790-00-0048	63790-00-0134	63790-00-0046
Component ID (Location)	MS-RV-5C	MS-RV-4D	MS-RV-1A	MS-RV-2B	MS-RV-1C
Date of Actuation (Mo/Da/Yr)	7/28/94	7/28/94	7/28/94	7/28/94	7/28/94
Time of Day (24 Hour Clock)	1408	1415	1441	1447	1452
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	≈15	≈15	≈15	≈15	≈15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	931	931	931	931	931
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	22 Sec.	34 Sec.	23 Sec.	11 Sec.	8 Sec.
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached?					

2.3 SUMMARY OF PLANT OPERATIONS

This report section is included in accordance with the guidance in Reg. Guide 1.16(C.1.b)

January 1994

- On January 14, 1994 the plant was maneuvered to a 49.8-hour scheduled power reduction for the purpose of conducting necessary maintenance on plant components. In addition, steam leaks and a condensate system pump were repaired.

- WNP-2 operated at near full power during the month.

February 1994

- WNP-2 operated at near full power during the month with no significant power reductions (greater than 20 percent). However, the reactor was in end-of-cycle coastdown with final feedwater temperature reduction in progress.

March 1994

- WNP-2 operated at near full power during the month with no significant power reductions (greater than 20 percent). However, the reactor was in end-of-cycle coastdown with final feedwater temperature reduction in progress.

April 1994

- On April 26, 1994, while operating at a reduced power level (approximately 50 percent), Control Room Operators observed slight fluctuations in the power range monitors. Accordingly, the reactor was manually scrammed as prescribed by the Abnormal Operating Procedures. During a followup assessment, it was determined that the cause was due to a malfunctioning valve transmitter in the Reactor Recirculation Flow Control System. The small changes in recirculation flow control valve position caused corresponding changes to reactor coolant flow and power parameters.

- On April 30, 1994 WNP-2 officially entered the R-9 1994 annual maintenance and refueling outage.

- WNP-2 operated near 65 percent power entering a forced outage on April 26, 1994 as a result of the manual scram. The annual refueling outage began on the last day of the month.

May

- WNP-2 was in the annual maintenance and refueling outage for the entire month.

June

- WNP-2 was in the annual maintenance and refueling outage for the entire month.

July

- WNP-2 was in the annual maintenance and refueling outage for almost the entire month.
- On July 30, 1994 WNP-2 officially concluded the annual maintenance and refueling outage and was ramping up to full power by the end of the month.

August

- WNP-2 was ramping up to full power at the beginning of the month.
- WNP-2 operated at near full power during the month.

September

- WNP-2 operated at full power during the month, except for minor downpower evolutions for periodic testing and maintenance.

October

- WNP-2 operated at full power during the month, except for minor downpower evolutions for periodic testing and maintenance.

November

- WNP-2 operated at full power during the month, except for minor downpower evolutions for periodic testing and maintenance.

December

- WNP-2 operated at full power during the month, except for minor downpower evolutions for periodic testing and maintenance.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

This information is provided in accordance with the guidance of Regulatory Guide 1.16, Section C.1.b(2)(e). In addition to safety-related equipment, components considered to be essential for power generation are also included.

2.4.1
Digital-Electro-Hydraulic (DEH) System

Elastomers in the DEH System were replaced due to seal/fluid incompatibility caused by improper flushing of the system. The problem was discovered when the plant experienced two, simultaneous DEH System pump failures.

2.4.2
Relay Replacement

A total of 52 Agastat relays were replaced due to a generic problem of deterioration of the nylon bobbin caused by prolonged exposure to high service temperature. The problem was discovered during investigation of a single Agastat relay that failed during surveillance testing.

2.4.3
Main Steam System

The existing General Electric (GE) Integrated Nuclear Measurement and Control (INMAC) Main Steam Line analog radiation monitors were replaced with GE Nuclear Measurement Analysis and Control (NUMAC) digital instruments. The replacement was performed due to the uncertain availability of future spares and past performance (i.e., output signal drift, spiking and inaccuracy) of the analog instruments.

2.4.4
Control Rod Drive System

The Scram Solenoid Pilot Valve (SSPV) diaphragms were replaced for all control rods following an investigation were it was revealed that the elastomer diaphragms exhibited generic and premature hardening and loss of flexibility. The problem was discovered when a control rod failed to insert upon receipt of the individual control rod scram signal from the toggle switches during surveillance testing.

2.4.5
Reactor Recirculation System

The position transmitters were replaced for the two Reactor Recirculation (RRC) System flow control valves. The replacement was performed to correct an oscillating flow control valve condition. The problem was a contributor to the scram the plant experienced just prior to entering the 1994 R-9 maintenance and refueling outage,

2.4.6
Containment Electrical Penetrations

A total of 10 electrical penetration modules were replaced, and 25 were reworked, that contained confirmed evidence of corrosion. Due to indications during the last operating cycle of moisture intrusion into a containment electrical penetration module containing non-safety related rod position indication system cables, other penetrations were examined to determine if a generic problem existed. As a result of investigation, a generic problem was discovered with containment electrical penetration modules using Scotchguard strain relief material with varglas wiring insulation.

2.5 FUEL PERFORMANCE

This section is provided in accordance with the requirements of the WNP-2 FSAR, Section 4.2.4.3, and Regulatory Guide 1.16, Revision 4, Section C.1.b.(4).

In accordance with commitments and requirements described in the WNP-2 FSAR, Section 4.2.4.3, a visual inspection of discharged fuel from Cycle 9 was performed in the month of November 1994. The purpose of the inspection was to verify assembly and fuel rod structural integrity. Although not a commitment, a visual inspection of two discharged fuel channels was also performed at the same time.

A total of eight fuel assemblies and two channels discharged at the end of Cycle 9 were inspected. No evidence of rod bow, abnormal fuel rod growth, debris, mechanical damage or offset tie rod latches was noted during the inspection of the assemblies. Furthermore, no significant nodular corrosion was observed. The surface of the clad was generally covered by uniform corrosion and crud, consistent with previous inspections.

The two fuel channels inspected displayed a uniform covering of light oxidation on unwelded surfaces. However, the heat-effected zones of the weld surface were not obscured by the oxidation. The heat-effected zones appeared cleaner than the unwelded surfaces of the channels. In addition, there was no apparent mechanical damage to the channels.

The results of these inspections are consistent with past experience and indicate that the bundles and channels are free of apparent, gross anomalies.

2.6 10CFR50.59 CHANGES, TESTS, AND EXPERIMENTS

Federal Regulations (10CFR50.59) and the Facility Operating License (NPF-21) allow changes to be made to the facility and procedures as described in the Safety Analysis Report, and tests or experiments to be conducted which are not described in the Safety Analysis Report without prior Nuclear Regulatory Commission (NRC) approval, unless the proposed change, test or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. In accordance with 10CFR50.59, summaries of the permanent design changes and temporary plant modifications completed in 1994 are provided. Included are summaries of the safety evaluations.

2.6.1 PLANT MODIFICATIONS

Permanent plant modifications at WNP-2 are implemented with a Plant Modification Request (PMR), Basic Design Change (BDC), Minor Design Changes (MDC), or Request For Technical Services (RFTS). The following PMRs/BDCs/MDCs/RFTSs implemented in 1994 required a Safety Evaluation in accordance with 10CFR50.59. Each permanent change was evaluated and determined neither to represent an Unreviewed Safety Question nor require a change to the WNP-2 Technical Specifications.

2.6.1.1
BDC 55-1564-0A

This BDC provided for top tier drawing corrections to show proper piping configurations of the Standby Gas Treatment deluge system.

It was concluded from the safety evaluation that no physical changes or field work was required. The design change was a document change only and was strictly administrative in context. This BDC did not increase the probability of Standby Gas Treatment system failure and, as a result, it would not increase the consequences of an accident evaluated previously in the Licensing Basis Documents (LBDs).

2.6.1.2
BDC 55-2256-0A

This BDC provided clarification concerning the use of Service Water valves SW-V-69A, SW-V-69B, SW-V-70A, and SW-V-70B during modes of operations other than shutdown.

The SW system is not credited with initiating any accident described in the LBDs. By allowing the valves be to opened during operation of the plant does not cause the inventory of the SW system to decrease. Compensatory measures, such as posting an operator at the valves, can be taken to ensure that the spray pond inventory is not depleted. Therefore, this activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.3
BDC 55-3035-0A

This BDC provided for revision of drawings to incorporate calculated bolt torque recommendations for motor-operators for valves LPCS-V-1 and RWCU-V-1.

It was concluded from the safety evaluation that implementation of this design change would not result in degradation of nuclear safety as described in the LBDs. The BDC only involves changes in recommended torques for motor-operator bolts, which are supported by calculations. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.4
BDC 83-0107-1B

This BDC provided for the implementation of the Filter/Polisher System for purifying Diesel Fuel Oil. This unit provides in-plant capability to maintain the quality of diesel fuel oil within the Technical Specification limits while the fuel is kept in prolonged storage.

It was concluded from the safety evaluation that installation of this design change would not increase the probability or the consequence of analyzed accidents as the Diesel Generator systems are mitigators and not initiators of accidents and maintain their reliability. The level of assurance that the Emergency Diesel Generator system will be available remains the same as if the fuel oil was transferred from another Emergency Diesel Storage Tank instead of from the Auxiliary Boiler Storage Tank.

2.6.1.5
BDC 87-0244-0J

This BDC provided for the removal of Reactor Recirculation Cooling (RRC loops A and B) and Residual Heat Removal (RHR Supply and Returns) piping system snubbers. This is part of a plant improvement program which focuses on enhancing pressure boundary integrity by eliminating failure prone mechanically complex pipe snubbers. Additional benefits are reduced radiological exposures due to maintenance and inspection.

It was concluded from the safety evaluation that this design change would not impact any system function or postulated design basis accident response. All piping system installations are qualified to the pertinent ASME Code and the WNP-2 design requirements. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.6
BDC 88-0393-1B

This BDC provided for the installation of the new LORAL Transient Data Acquisition System (TDAS) in the Main Control Room to enhance monitoring capabilities.

It was concluded from the safety evaluation that this design change would not impact any system function or postulated design basis accident conditions. The installed TDAS equipment is Seismic Category 1 and does not directly interface with any safety-related equipment or systems. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.7
BDC 89-0151-0B

This BDC provided for addition of drain lines connecting to condenser return lines at two locations: 1) the low point of the seal steam piping on the Reactor Feed Water (RFW) turbine skid; and 2) the low point of the RFW turbine exhaust casing. This design change was made on both the A and B sides.

It was concluded from the safety evaluation that this design change would not affect the function or form of any of the existing components. An unreviewed safety question is not introduced as a result of this change. The new drain lines will improve the reliability of the RFW turbine by reducing erosion caused by condensate accumulation. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.8
MDC 90-0226-0A

This MDC provided follow up documentation for the removal of Reactor Water Clean Up (RWCU) relief valve RWCU-RV-2 which prevented a hot startup of the system. Field changes were completed by means of the Temporary Modification process which removed RWCU-RV-2 and installed a blind flange onto the pressure piping.

It was concluded from the safety evaluation that this design change has decreased the probability of an LBD accident due to the improved integrity of the boundary by the reduction of potential primary coolant containment devices.

2.6.1.9
BDC 90-0346-0A

This BDC provided for the changes to the drive ratio of RCIC-MO-8, slowing it to provide greater closing torque to meet degraded voltage system requirements. The design changes consisted of the replacement of the motor pinion gear, worm shaft gear, and the torque spring pack to accommodate the greater thrust associated with the gear ratio change.

It was concluded from the safety evaluation that this design change does not change the fit or basic form of any of the existing components. An unreviewed safety question is not introduced as a result of this change. The existing motor operator will be improved by a gear change-out that provides a higher available thrust. The increase in the HELB environmental profile temperatures has been evaluated and approved. All equipment required for safe operation of the plant will remain qualified. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.10
BDC 91-0422-0A

This BDC provided for the removal of the internals from three check valves which are not required and which must be functionally tested quarterly to be in compliance with the Inservice Testing (IST) Program (Generic Letter 89-04). These valves were provided generically to prevent back flow between RHR pump minimum flow lines. These valves are not required because there is no connection between the lines as each minimum flow circuit is routed independently to discharge into the wet well.

It was concluded from the safety evaluation that the removal of the valve internals would assure flow with no affect on the pressure boundary integrity thereby reducing the probability of an accident. No new failure modes are introduced, therefore, there is no increase in the probability of malfunctions of equipment important to safety and the activity would not create the possibility of an accident of a different type than any evaluated in the LBD.

2.6.1.11
BDC 92-0095-0A

This BDC provided for the increased capacity for 125 volt Battery, E-B1-1, by replacing the existing cell units with the next larger size.

It was concluded from the safety evaluation that the safety function of the battery remains unchanged after this modification. Since the battery needs replacement, replacing the cells with the slightly higher capacity cell units is a way to increase the battery's capacity margin for supplying emergency DC power during postulated events. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.12
BDC 92-0095-1B

This BDC provided for the increased capacity for 125 volt Battery, E-B1-2, by replacing the existing cell units with the next larger size.

It was concluded from the safety evaluation that the safety function of the battery remains unchanged after this modification. Since the battery needs replacement, replacing the cells with the slightly higher capacity cell units is a way to increase the battery's capacity margin for supplying emergency DC power during postulated events. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.13
BDC 92-0158-0A

This BDC provided for the modification of the Reactor Heat Removal (RHR) injection valve control logic by installing keylock bypass switches to allow throttling of valves RHR-V-42A, RHR-V-42B, and RHR-V-42C in accordance to Emergency Operating Procedures (EOPs). When the keylock is in the bypass position it will override the logic interlock signals and the seal-in function to allow operators to more readily carry out EOP-required actions.

It was concluded from the safety evaluation that there was no impact to LBD analysis. The use of keylocked switches to override the valve logic will increase the probability that the valves will perform their intended function as required by the EOPs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.14
BDC 92-0158-0B

This BDC provided for the modification of High Pressure Core Spray (HPCS) injection valve control logic by installing keylock bypass switches to allow throttling of valve HPCS-V-4 in accordance with Emergency Operating Procedures (EOPs).

It was concluded from the safety evaluation that there was no impact to LBD analysis. The use of keylocked switches to override the valve logic will increase the probability that the valves will perform their intended function as required by the EOPs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.15
BDC 92-158-0C

This BDC provided for the modification of Low Pressure Core Spray (LPCS) injection valve control logic by installing keylock bypass switches to allow throttling of valve LPCS-V-5 in accordance with Emergency Operating Procedures (EOPs).

It was concluded from the safety evaluation that there was no impact to LBD analysis. The use of keylocked switches to override the valve logic will increase the probability that the valves will perform their intended function as required by the EOPs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.16
BDC 92-158-0D

This BDC provided for the installation of keylock bypass switches to allow the Main Steam Isolation Valves (MSIVs) to be opened, in accordance with Emergency Operating Procedures (EOPs), while low vessel level and high steam tunnel temperature isolation conditions exists.

It was concluded from the safety evaluation that there was no impact to LBD analysis. The use of keylocked switches to override the valve logic will increase the probability that the valves will perform their intended function as required by the EOPs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.17
BDC 92-0178-5F

This BDC provided for the installation of new low dead band pressure switches to replace HPCS Diesel Starting Air (DSA) switches DSA-PS-15 and DSA-PS-16. The system design function is to ensure that the HPCS diesel is allowed three starts in accordance with FSAR commitments.

It was concluded from the safety evaluation that the pressure switch replacements made by this BDC do not adversely affect the safety or performance of any plant system. DSA-PS-15 and DSA-PS-16 are Quality Class 1, and serve a passive safety-related function which is to maintain the system pressure boundary. It was also concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.18
BDC 92-0178-6G

This BDC provided for the ASME, Section VIII, re-rating of HPCS DSA System air receivers. The purpose of this design change was to upgrade air receivers DSA-AR-1C and DSA-AR-2C to a higher working pressure and provide direction for changing the setpoint pressures of the associated relief valves and low pressure alarms.

It was concluded from the safety evaluation that there is no impact to the LBD analyses. The HPCS is an accident mitigation system. There are no credible mechanisms where this modification could cause an inadvertent operator error to start HPCS injection. Furthermore, there are no new mechanisms which could cause automatic start of HPCS. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.19
BDC 92-0297-0A

This BDC provided for the replacement of the existing triangular refueling mast with a General Electric NF-500 tubular refueling mast supplied with a video camera.

It was concluded from the safety evaluation that the consequences of a fuel handling accident are unaffected by NF-500 refueling mast design. Therefore, this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.20
<u>BDC 92-0220-0A</u>

This BDC provided for the Turbine Generator (TG) Building exhaust fan changeout. The TG fan motors on TG-F-1A, TG-F-1B, TG-F-1C, and TG-F-1D were changed from 100 hp to 200 hp to improve the system's ability to control temperature and handle the added heat load expected from Reactor Power Uprate.

It was concluded from the safety evaluation that the TG building exhaust fan system has no safety-related function and all equipment affected by this modification is Class 2. Therefore, implementation of this design change has been determined not to result in a degradation of nuclear safety as described in the LBDs. In addition, it was concluded that this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.21
<u>BDC 92-0223-1A</u>

This BDC provided for the addition of scram discharge volume vent and drain valve close signals to the plant process computer. The objective of this design change was to eliminate unnecessary and excessive radiation dose by providing remote indication to the plant process computer for the Scram Discharge Volume (SDV) vent and drain valves CRD-V-10, CRD-V-11, CRD-V-180, and CRD-V-181.

It was concluded from the safety evaluation that this design change does not affect LBD analysis. The open and close limit switch on each SDV valve provides position indication only in the control room by means of indicating lights and a plant computer point for the open position, and the close position after this change is implemented. Neither the SDV valves nor their respective limit switches contribute to or prevent the scram function. They also are not assumed to be the initiator of an accident. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.22
<u>BDC 92-0286-0A</u>

This BDC provided for the substitutions of Containment Instrument Air (CIA) pressure regulating valves. The existing nitrogen bottle high pressure regulators and associated isolation valves for CIA-TK-1A through CIA-TK-15A, CIA-TK-20A, and CIA-TK-1B through CIA-TK-20B are designed to regulate pressure from the nitrogen cylinders in the railroad bay of the reactor building, plus two in the diesel generator building. The existing components for the regulator/isolation assembly are no longer manufactured and are obsolete. This design change replaces these assemblies with the manufacturer's equivalent components. The new components

are slightly different in form and, as a result, require minimal modifications to the lower hanger support.

It was concluded from the safety evaluation that this design change will not result in a degradation of nuclear safety as described in the LBDs. This design change will result in neither an increase nor a decrease in the reliability or safety of the CIA system as the flow characteristics of the new regulators are equivalent to the existing pressure regulators. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.23
<u>BDC 92-0301-0A</u>

This BDC provided for the realignment of Standby Gas Treatment (SGT) valves SGT-V-2A AND SGT-V-2B from normally closed to normally open. This design change realigns SGT-V-2A and SGT-V-2B from normally closed/fail open to normally open/fail open, which eliminates the possibility of a system failure as a result of these valves failing to open on demand.

It was concluded from the safety evaluation that this design changes does not impact LBD analysis. This change supports two-train operation of the SGT system required for post-LOCA secondary containment drawdown. The possibility of a system failure resulting from these valves failing to open on demand is eliminated by these valves being placed in the normally open position. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.24
<u>BDC 93-0093-0B</u>

This BDC provided for the change from an existing demineralized water system located in the basement of the General Service Building to a contract-supplied system to be located in a trailer north of the plant near the water treatment building.

It was concluded from the safety evaluation that this design changes has no impact on any safety-related system. The demineralized water processing function is not safety-related and the system does not interact with any safety-related systems. The result of the modification is the production of higher quality demineralized water. In addition, it was concluded that this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.25
BDC 93-157-0A

This BDC provided for the replacement of four Containment Supply Exhaust/Purge (CEP/CSP) valves to improve seat leakage. Existing valves CSP-V-3, CSP-V-4, CEP-V-1A, and CEP-V-2A were unreliable and experienced difficulty in passing seat leakage testing.

It was concluded from the safety evaluation that this design change has no effect on the safety function of these valves or the system. The new valves have the same closure time, leak tightness, quality and seismic requirements as the existing valves. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.26
BDC 93-0227-0A

This BDC provided for the installation of lead shielding around level switches on the CRD Scram Discharge Volume (SDV). The shielding was requested for level switches CRD-LS-13C through CRD-LS-13H to reduce the general work area dose rates.

It was concluded from the safety evaluation that this design change would not adversely affect the design function of these components or the SDV system. The shielding lead sheet at each level switch will be installed to withstand Seismic Class I load conditions without failure. The existing support at each level switch is adequate for the additional load of shielding and its support steel or lead sheet for all load cases. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.27
BDC 93-0251-0A

This BDC provided for the removal of the electric brakes from fourteen SGT valve actuator motors. These brakes are removed because the motor-operators on these valves have "locking" gear ratios and the brakes are not required.

It was concluded from the safety evaluation that this design change would not adversely affect the design function or performance of the system's safety function. Brakes are used to minimize the inertial loads that may occur after motor de-energization, or to maintain a lock on the motor shaft once the motor has been de-energized. Limitorque Maintenance Update 92-2 concluded that motor brakes have "an insignificant affect on the inertial overshoot...", the only reason to continue using brakes is to compensate for a "non-locking" gear ratio in the valve operator. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.28
BDC 93-0307-0A

This BDC provided for the decrease of the DC inductive loading on Potter & Brumfield MDR relays in DG-1 and DG-2 starting air control circuits DG-RLY-DG1/K16 and DG-RLY-DG2/K16. This design change is in response to NRC Information Notice 92-19 which indicated that Potter & Brumfield non-latching rotary relays of the "MDR" series are susceptible to failure when switching excessive inductive loads.

It was concluded from the safety evaluation that this design change had no impact to plant operations or response capability to accident monitoring. This design change will preserve the current diesel air start system function and will not jeopardize the ability of the diesel engines to start on demand. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.29
BDC 94-0022-0A

This BDC provided for the installation of a bypass around RHR-V-6A. This will allow acceptable leakage past RHR-V-8 and RHR-V-9 to be returned to the suppression pool.

It was concluded from the safety evaluation that this design change will prevent pressurization of the shutdown cooling suction line without affecting the operation of either RHR Loops A or B in the Shutdown Cooling (SDC) mode. The change itself will introduce no new accident or increase the consequences of any accident previously analyzed.

2.6.1.30
BDC 94-0043-0A

This BDC provided for the modification of pressure locking susceptible valves RHR-V-8, RHR-V-9, and LPCS-V-5. This design change will permanently make these valves unidirectional, i.e. the reactor side valve disk/seat will no longer provide isolation capabilities. However, since all the flow path isolation conditions, including leak testing, only require the low pressure (non-reactor) side seat/wedge to seal, this change will still allow for proper isolation.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The ability of these valves to open will be enhanced by these design changes. The original design function of these valves to open will be enhanced by these design changes. Valve close functions are not affected. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.31
BDC 94-0048-0A

This BDC provided for the replacement of containment penetration X101B modules. The replacement of these modules was needed in order to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement modules are identical in form, fit, and function to the existing modules in their role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.32
BDC 94-0048-1B

This BDC provided for the replacement of containment penetration X101A module position 1. The module required replacement to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement module is identical in form, fit, and function to the existing module in its role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.33
BDC 94-0048-2C

This BDC provided for the replacement of containment penetration X101C module position 3. The module required replacement to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement module is identical in form, fit, and function to the existing module in its role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.34
BDC 94-0048-3D

This BDC provided for the replacement of containment penetration X101D module position 3. The module required replacement to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement module is identical in form, fit, and function to the existing module in its role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.35
BDC 94-0071-0A

This BDC provided for the removal of the torque switch from Auxiliary Steam (AS) valves AS-V-68A and AS-V-68B open circuit. This design change allows the subject valve opening to be controlled by limit switch instead of torque switch.

It was concluded from the safety evaluation that this design change would not result in degrading nuclear safety as described in the LBDs. Since valve AS-V-68A and AS-V-68B do not have an "open" safety function, removing the "torquing open" travel limit does not reduce the valves ability to complete their respective "close" safety function. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.36
BDC 94-0072-0A

This BDC provided for Reactor Core Isolation Cooling (RCIC) System valve motor-operator RCIC-MO-76 gear/spring pack changeout. This design change has increased available operator thrust by changing out RCIC-MO-76 motor pinion gear, worm shaft gear and spring pack.

It was concluded from the safety evaluation that implementation of this change has not resulted in a degradation of nuclear safety as described in the LBDs. The basic functions of the RCIC system, RCIC-V-76 and RCIC-MO-76 are unchanged. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.37
BDC 94-0083-0A

This BDC provided for the replacement of Containment Air Monitors System (CMS) charcoal filter. The replacement of the charcoal filters in sample racks CMS-SR-20 and CMS-SR-21 will reduce oxygen leaks into the containment.

It was concluded from the safety evaluation that the drywell atmosphere monitoring system is not safety-related, has no safety function nor will its failure jeopardize safety/safety-related equipment. In addition, it was concluded that this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.38
BDC 94-0102-0A

This BDC provided for the redesign of the steam tunnel access security hatch. This design change provided access into the steam tunnel which can be secured against unauthorized entry into the vital area without subjecting personnel to unnecessary levels of radiation exposure while removing the security barrier.

It was concluded from the safety evaluation that this design change does not constitute a significant change to the form, fit or function of the access hatch or the blow-out panel on which it is attached. An unreviewed safety question is not introduced as a result of this change. The design change does not degrade the ability of the panel to perform its safety function, and the access panel itself will not become a projectile under any design conditions that could damage any equipment important to safety. This modified hatch satisfies all requirements as a security boundary to the vital area. In addition, it was concluded that this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.39
BDC 94-0137-0A

This BDC provided for the removal of the backseat ring on SW-V-44.

It was concluded from the safety evaluation that removing the backseat ring will not affect the ability of the valve to perform its function. Any leakage that might result through the packing gland rings would be minor and would not affect operation of the plant. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.40
<u>BDC 94-0139-0A</u>

This BDC provided for the replacement of containment penetration X104A module position 1. The module required replacement to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement module is identical in form, fit, and function to the existing module in its role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.41
<u>BDC 94-0143-0A</u>

This BDC provided for the replacement of Main Steam (MS) motor operator MS-MO-19. The existing operator gear rating of 90 ft-lbs and motor start torque of 5 ft-lbs are inadequate. Replacement will increase the gear rating and motor start torque.

It was concluded from the safety evaluation that no reduction in safety will occur as a result of this change. The design change does not change the basic fit or form of any of the components. The safety functions and operational parameters are unchanged. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.42
<u>BDC 94-0164-0A</u>

This BDC provided for the replacement of containment penetration X105A module position 1. The module required replacement to correct low/high insulation resistance values for installed circuits.

It was concluded from the safety evaluation that this design change would have no impact involving safety-related function or design of the containment penetrations. The replacement module is identical in form, fit, and function to the existing module in its role as a containment boundary. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.43
BDC 94-0175-0A

This BDC provided for the sealing of the downstream piping of Reactor Recirculation (RRC) System Loop B drain valves RRC-V-51B and RRC-V-52B. These valves were leaking by their seats and cannot be disassembled for repair. The addition of caps downstream is to assure identified leakage to the drywell sump does not become an operational concern during the next operating cycle.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The design change does not change the basic fit or form of any of the components. The safety functions and operational parameters are unchanged. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.44
BDC 94-0183-0A

This BDC provided for the replacement of penetration X-105C module positions 1 and 2 with blank modules. Removal of spare modules are required for electrical examination and testing for evaluation of possible module electrical degradation.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Circuits passing through containment penetration X-105C module positions 1 and 2 are spares. Therefore, the penetration modules have no safety-related function other than the containment pressure boundary function, which is unaffected. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.45
BDC 94-0192-0A

This BDC provided for vent holes and cover plates for wetwell electrical terminal boxes TB-C500 and TB-C501 (Penetrations X-107A and X-107B).

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. This design change has no impact on any equipment operating mode, nor are the consequences of any accident scenario changed by the inclusion of these structures.

2.6.1.46
BDC 94-0197-0A

This BDC provided for the replacement of penetrations X-100B-1 and X-100D-2. Removal of spare modules is required for electrical examination and testing for evaluation of possible module electrical degradation.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Circuits passing through containment penetrations X-100B-1 and X-100D-2 module positions 1 and 2 are spares. Therefore, the penetration modules have no safety-related function other than the containment pressure boundary function, which is unaffected. As a result, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.47
BDC 94-0198-0A

This BDC provided for the replacement of penetration X-107A. Removal of spare module is required for electrical examination and testing for evaluation of possible module electrical degradation.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Circuits passing through containment penetration X-107A. Therefore, the penetration modules have no safety-related function other than the containment pressure boundary function, which is unaffected. As a result, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.48
BDC 94-205-0A

This BDC provided for the relocation of the pressure regulators for CRD-V-180 and CRD-V-181. This change will provide assurance that these valves will close upon receipt of a signal from a reactor scram.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. This design change does not change the function of any of the components. The new pressure regulator will perform the same function as the old one. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.49
BDC 94-0241-0A

This BDC provided for the change of power supply for Suppression Pool Temperature Monitoring (SPTM) instruments required for Division II Alternate Remote Shutdown to essential power supply E-E/S-299 in Control Room panel E-CP-H13/P833.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Rather than providing a fuse for cable CBDA-28/BMISC-9737, the design change will supply the Appendix R SPTM loads from a power supply that is not affected by the design basis fire. These loads will be supplied from E-E/S-299 in the same panel. No other cables supplied from this power supply are routed through unprotected raceways external to the control room. This design change does effect safety-related equipment other than these SPTM instruments. In addition, it was concluded that this design change will not impact the consequences of an accident evaluated previously in the LBDs.

2.6.1.50
RFTS 93-01-053

This RFTS added weld details to the Sheet Metal Standard, Certified Vendor Information (CVI), CVI 216-00.1658.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The design intent of the system is not altered in any way by this change. The change provides direction for welding various duct components and ducts. It also provides minimum access panel sizes and permits the use of stainless steel in the construction of HVAC ducts. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.2 TEMPORARY MODIFICATIONS AND INSTRUMENT SETPOINT CHANGES

The following are summaries of temporary modifications, instrument setpoint changes and motor-operated valve setpoint changes. As required by 10CFR50.59, each change was evaluated and determined neither to represent an Unreviewed Safety Question nor a change to the WNP-2 Technical Specifications. Temporary modifications are made by means of the Temporary Modification Request (TMR) process, instrument setpoint changes are made under the Instrument Setpoint Change Request (ISCR) process and motor-operated valve setpoint changes are made under the MOV Setpoint Change Request (MSCR) process.

2.6.2.1
TMR 94-010

This TMR provided for the addition of heat trace to Reactor Building Electrical Containment Penetration X101B to prevent deterioration of the penetration.

It was concluded from the safety evaluation that the installation of the temporary heat trace would not increase the consequences of an accident evaluated previously in the Licensing Basis Documents (LBDs) because the penetration seal temperature limit of 158 degrees fahrenheit would not be exceeded should the temperature switch fail.

2.6.2.2
TMR 94-012

This TMR provided for the installation of a temporary pressure gauge on the low side of Residual Heat Removal (RHR) System Differential Pressure Indicating Switch RHR-DPIS-12B to provide for a means to measure pressure between valves RHR-V-8 and RHR-V-9/209.

It was concluded from the safety evaluation that the modification would not increase the consequences of an accident evaluated previously in the LBDs. The installation, use, and possible failure of the temporary pressure gauge components are bounded by the consequences already analyzed in the LBDs for the loss of the RHR shutdown cooling suction piping.

2.6.2.3
TMR 94-013

This TMR provided for the installation of a Probe Mux Interface Card for Control Rods 46-31 and 58-23 to restore control rod position indication by accurately monitoring the status of the control rod reed switches in the rod position indicating probe and converting the information to the Rod Position Indication System electronics.

It was concluded from the safety evaluation that the modification would not increase the consequences of an accident evaluated previously in the LBDs because the method for processing rod position indication information through the Rod Position Indication System electronics was not changed.

2.6.2.4
TMR 94-014

This TMR provided for the installation of a Probe Mux Interface Card for Control Rod 46-15 to restore control rod position indication by accurately monitoring the status of the control rod reed switches in the rod position indicating probe and converting the information to the Rod Position Indication System electronics.

It was concluded from the safety evaluation that the temporary modification would not increase the consequences of an accident evaluated previously in the LBDs because the method for processing rod position indication information through the Rod Position Indication System electronics was not changed.

2.6.2.5
TMR 94-017

This TMR provided for defeating of the automatic load shed on an "FA" accident signal for Circulating Water (CW) System Pump CW-P-1C with CW-P-1A out-of-service and a caution tag placed at the control switch. The caution tag was to warn operators not to operate CW-P-1A until the "FA" trip is reinstated for CW-P-1C.

It was concluded from the safety evaluation that the proposed activity would not increase the probability of occurrence of an accident evaluated previously in the LBDs because the trip of the CW pumps is an accident mitigation feature. With CW-P-1A out-of-service and starting of the pump administratively controlled, removing the "FA" trip from CW-P-1C could not result in an increase in the probability of an accident.

2.6.2.6
<u>TMR 94-018</u>

This TMR provided for the addition of an ion exchange column in the Reactor Closed Cooling (RCC) System to control conductivity excursions and decrease the possibility of corrosion occurring in the system.

It was concluded from the safety evaluation that the modification would not increase the consequences of an accident evaluated previously in the LBDs because the TMR boundary was located outside the containment isolation valves. It was determined that accidental resin intrusion from the temporary ion exchange column and subsequent isolation valve blockage was not a credible event. Therefore, the RCC containment isolation function was not affected by the installation of this modification.

2.6.2.7
<u>TMR 94-043</u>

This TMR provided for bypassing of the manual isolation contacts in the Primary Containment Isolation Logic to allow for replacement of HFA relays during a Reactor Protection System (RPS) bus power outage during the 1994 R-9 maintenance and refueling outage . The automatic isolation features remained operable.

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs because the temporary modification would not impact the systems ability to achieve automatic isolation in the event of an accident. Furthermore, isolation valve control was still available from the Main Control Room to serve as a backup to the automatic isolation system.

2.6.2.8
<u>TMR 94-046</u>

This TMR provided for the addition of heat trace to piping for Reactor Building Electrical Containment Penetrations X101A, X101C, X101D, X102A, X105D, X104C, X104D, X100A, X100B, X100C, X100D, X107A and X107B to prevent deterioration of the penetration.

It was concluded from the safety evaluation that the installation of the temporary heat trace would not increase the consequences of an accident evaluated previously in the LBDs because the penetration seal temperature limit of 158 degrees fahrenheit would not be exceeded should the temperature switch fail.

2.6.2.9
TMR 94-053

This TMR provided for the installation of jumpers to bypass Radwaste Building HVAC Moisture Sensors WOA-MS-54A and WOA-MS-54B to allow Heaters WOA-EHC-54A and WOA-EHC-54B to remain operable and maintain inlet air to the filter unit charcoal at 70 percent or less relative humidity.

It was concluded from the safety evaluation that the installation of the temporary modification would not increase the consequences of an accident evaluated previously in the LBDs because the activity could not jeopardize operability of the Control Room Emergency Filtration System. By limiting the temporary modification to less than 80 degrees farhenhiet outside temperature, there would be no reduction in the analyzed control room habitability.

2.6.2.10
ISCR 1219

This ISCR provided for changing of the electronic overspeed trip setpoint from 113.3 percent to 121 percent of rated load speed for the Reactor Core Isolation Cooling (RCIC) System turbine to reflect a recommendation from General Electric.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident evaluated previously in the LBDs. Raising the electronic overspeed trip increases the reliability of the RCIC System by preventing spurious trips during system initiation.

2.6.2.11
MSCR 567 and 644

These MSCRs provided for decreasing Residual Heat Removal (RHR) System valve motor-operator RHR-MO-27A differential pressure in both directions by a small amount. The minimum setpoint had increased due to the use of a larger mean seat diameter, higher valve factor, rate of loading and stem factor degradation.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident evaluated previously in the LBDs because use of these new maximum and differential pressures provide increased assurance that the new setpoint is appropriate for the motor-operated valve.

2.6.2.12
MSCR 675, 676 and 678

These MSCRs provided for increasing Main Steam (MS) System valve motor-operator MS-MO-67A open bypass setting from 10-25 percent open to 40-50 percent open (or 20 percent open flowpath) to ensure that the valve opens and has a minimum torque switch setting to minimize backseating thrust loads.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident evaluated previously in the LBDs. A minimum open torque switch setpoint reduces the open thrust loads at the end of the stroke and ensures that the MOV will open under the maximum expected differential pressure conditions after torque switch bypass.

2.6.2.13
MSCR 712

This MSCR provided for increasing Residual Heat Removal (RHR) System valve motor-operator RHR-MO-8 design basis differential pressure, for the open direction, from 75 psid to 161 psid based on a review of the design basis and setpoint calculations.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident evaluated previously in the LBDs because increasing the pressure for opening results in a conservatively higher calculated thrust to open the valve. Furthermore, it was concluded that the change provides assurance of valve operation at design basis conditions.

2.6.2.14
MSCR 713

This MSCR provided for increasing Residual Heat Removal (RHR) System valve motor-operator RHR-MO-9 design basis differential pressure, for the open direction, from 75 psid to 161 psid based on a review of the design basis and setpoint calculations.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident evaluated previously in the LBDs because increasing the pressure for opening results in a conservatively higher calculated thrust to open the valve. Furthermore, it was concluded that the change provides assurance of valve operation at design basis conditions.

2.6.2.15
MSCR (Several: Safety Evaluation 94-54)

These MSCRs provided for changing of the Motor-Operated Valve Master Data Sheets for several motor-operated valves.

In all cases, it was concluded from the safety evaluation that these activities would not increase the probability of occurrence of an accident evaluated previously in the LBDs. The MOV Master Data Sheets convey necessary information from design basis calculations allowing technicians to properly set the motor actuator torque and limit switches to ensure operability at design basis conditions, without exceeding maximum thrust or torque limits or maximum torque capability under degraded voltage conditions. Typically, motor-operated valves mitigate the effects of design basis accidents. Therefore, establishing the torque and/or thrust setpoint would not cause an accident evaluated previously in the LBDs.

2.6.2.16

MSCR (Several; Safety Evaluation 94-55)

These MSCRs provided for changing of the Motor-Operated Valve Master Data Sheets for several motor-operated valves.

In all cases, it was concluded from the safety evaluation that these activities would not increase the probability of occurrence of an accident evaluated previously in the LBDs. The MOV Master Data Sheets convey necessary information from design basis calculations allowing technicians to properly set the motor actuator torque and limit switches to ensure operability at design basis conditions, without exceeding maximum thrust or torque limits or maximum torque capability under degraded voltage conditions. Typically, motor-operated valves mitigate the effects of design basis accidents. Therefore, establishing the torque and/or thrust setpoint would not cause an accident evaluated previously in the LBDs.

2.6.3 FSAR CHANGES

General changes to the FSAR evaluated within the definition of 10CFR50.59 are reported in this section. FSAR changes are processed through the SAR Change Notice (SCN) process.

2.6.3.1 SCN 93-081

This SCN provided for revision of the Diesel Generator loading schedules due to revised calculations. The diesel loading tables require revision upon loading changes.

It was concluded from the safety evaluation that revision of the loading schedules would not impact the consequences of an accident or reduce evaluation for the margin of safety. The diesel generators are accident mitigating features and, as such, cannot cause an accident. The calculations were revised to reflect current diesel loading and it was verified that the diesel generators would continue to perform their intended safety functions. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.3.2 SCN 93-120

This SCN provided for the correction of an FSAR discrepancy pertaining to the pilot solenoids for each Main Steam Isolation Valve (MSIV). The FSAR was to be revised to correctly show that the MSIV pilot valves are powered from Reactor Protection System Motor-Generator Sets A and B.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. No field work was required. The function of the MSIVs is to mitigate the effects of an accident. The proposed change would not increase the consequences of malfunction of equipment important to safety. There is no change to the isolation signals for the MSIVs as a result of this activity. Furthermore, the valves still function to mitigate events and postulated malfunctions are unchanged. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.3.3
SCN 94-001

This SCN provided for the clarification and update of the solid waste management system description in the FSAR.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Furthermore, the changes were primarily editorial or clarifying in nature and would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.3.4
SCN 94-006

This SCN provided for the correction of Table 3.9-3 "ASME Class 2 and 3 Active Pumps and Valves" by removing check valve RHR-V-89. This valve is located on the RHR-SW cross tie line and is installed in the RHR "B" heat exchanger room.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Table 3.9-3 identifies all "active" ASME class 2 and 3 components. RHR-V-89 performs a "passive" safety function. Changing the classification of RHR-V-89 from active to passive will not increase the probability of occurrence of an accident evaluated previously in the LBDs. The valve is not required to change state to perform its safety-related function.

2.6.3.5
SCN 94-040

This SCN provided for the revision of sections of the FSAR to show that the post LOCA recombiner is used for containment oxygen control instead of hydrogen control, and includes other Containment Atmosphere Control (CAC) System changes to stipulate the minimum performance requirements to meet the required oxygen control limits.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. This activity serves to better define the purpose and requirements of the CAC System. Because the CAC System is used for accident mitigation and cannot in itself cause an accident, there will be no increase in the probability of an accident evaluated previously in the LBDs. Furthermore, because the CAC System meets or exceeds the minimum required flow rate necessary to maintain oxygen level in containment less than 4.8 percent, there will be no increase in the consequences of an accident evaluated previously in the LBDs.

2.6.3.6
SCN 94-052

This SCN provided for inclusion of the description of the 12 volt battery used for starting diesel fire pump FP-P-110.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The diesel fire pump is not considered in any accident evaluation. Furthermore, there is no mechanism where operation of the diesel fire pump can adversely affect reactivity controls or increase the probability of occurrence of an accident evaluated previously in the LBDs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.3.7
SCN 94-059

This SCN provides for the correction to FSAR Table 6.2-16 and Figures 6.2-31f and 6.2-31p to show RHR-V-46A, RHR-V-46B, and RHR-V-46C no longer have an inline check valve.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. It was also concluded from the safety evaluation that there is no mechanism that links the check valves and initiating events of accidents. Therefore, this activity would not increase the probability of occurrence of an accident previously evaluated in the LBDs.

2.6.3.8
SCN 94-061

This SCN provides for clarification of Appendix R operator actions credited, improved user implementation, and identifying time constraints for credited operator actions.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Clarification is provided to address control room evacuation and ensure the evacuation and subsequent actions are accomplished within the limits of the safe shutdown analysis. The changes to the FSAR ensure that the credited Appendix R operator actions necessary to successfully achieve safe shutdown are completed in the proper sequence, location and time constraint. These changes provide additional clarification to ensure that operator actions necessary for fire mitigation are accomplished. Therefore, this activity would not increase the probability of occurrence of an accident previously evaluated in the LBDs.

2.6.3.9
SCN 94-066

This SCN provided for the revision of Chapter 15.6.2 to include revised calculational results for the Instrument Line Break Accident (ILBA). The revision is based on a 1/2" restricting orifice rather than the analyzed 1/4" orifice (1/2" and 3/8" orifices are installed in the plant).

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. A re-analysis of the ILBA was performed to evaluate the impacts of mass release and dose to the public based on the actual installed orifices in the instrument lines. The dose to the public for the ILBA will remain bounded by the values documented in the WNP-2 SER. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.4 PROBLEM EVALUATIONS

The Plant Problems-Plant Problem Reports Procedures (PPMs 1.3.12 and 1.3.12A) provide instructions for the disposition and documentation of plant problems. Plant problems are documented on a Problem Evaluation Request (PER). The following PERs were evaluated to provide assurance that the disposition did not involve an Unreviewed Safety Question or represent a change to the Technical Specifications.

2.6.4.1
PER 293-1320

This PER documents the discrepancies in the response time testing conducted to fulfill the requirements of Technical Specification Surveillance Requirement 4.3.2.3 for Isolation Actuation Instrumentation. The deficiency is the lack of overlap between the isolation logic and the component actuation circuits. Various PPMs for APRM Flow Biased Thermal High-High Trip Channel surveillances were temporarily cancelled as a result of this PER.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced due to the cancellation of the surveillances. There is no discernable increase in the probability of a design basis accident or operational transient. The cancellation of the procedures have only an administrative impact and will be re-initiated before the Technical Specification commitment requires they perform the procedures. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.4.2
PER 293-1411

This PER documented a condition in the EDG air start control logic where the connected load on contacts AB and DE of relay DG-RLY-DG1/K16 exceeds vendor recommended contact rating for switching inductive loads. This condition will affect contact life which has potential impact on long term reliability of the EDG control logic.

It was concluded from the safety evaluation that EDG operation until the 1994 R9 maintenance and refueling outage, with new K16 relays in the existing control circuit design, would not change its safety function. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.4.3
PER 294-0043

This PER documents the continuous pressurization of the RHR Shutdown Cooling Suction piping. This is attributed to internal leakage past containment isolation valves RHR-V-8 and RHR-V-9/RHR-V-209. This is related to BDC 94-0022-OA which provided for the installation of a bypass around RHR-V-6A. This will allow acceptable leakage past RHR-V-8 and RHR-V-9 to be returned to the suppression pool.

It was concluded from the safety evaluation that this design change will prevent pressurization of the shutdown cooling suction line without affecting the operation of either RHR Loops A or B in the Shutdown Cooling (SDC) mode. The change itself will introduce no new accident or increase the consequences of any accident previously analyzed.

2.6.4.4
PER 294-0054

This PER documents the installation of incorrectly sized orifices on the instrument lines penetrating primary containment. Rather than the analyzed 1/4" orifices, the one-inch lines use a 1/2" orifice and the 3/4" lines use a 3/8" orifice.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this condition. The orifices used in the instrument lines penetrating primary containment are not an initiator for any accident previously evaluated. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.4.5
PER 294-0416

This PER documents the out of round lift ring on the hanger arm of LPCS-V-6.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this condition. The air operated lift feature had no safety related function. The ring for lifting is out of round but does not alter the valve type, operating time, failure position, valve size, design temperature, or design pressure. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.4.6
PER 294-0433

This PER documents the acceptance of seven existing shims under the yoke of RHR-V-73A based on vendor's approval. Vendor's written concurrence was conditional on proper valve operation, which has been proven multiple times over the last ten years by means of the LLRT/IST Testing/MOV Diagnostic Testing.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this condition. Shimming was installed as necessary to achieve proper motor operator position. Furthermore, it was concluded that the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.5 PLANT TESTS AND EXPERIMENTS

This section of the report covers WNP-2 Plant tests and experiments not described in the Safety Analysis Report as required by 10CFR50.59.

There were no tests or experiments performed under the provisions of 10CFR50.59 in 1994.

2.6.6 PLANT PROCEDURE CHANGES

The Plant Procedure Control Program requires a 10CFR50.59 evaluation whenever a procedure is changed. This provides assurance that the change does not require a change to the Technical Specifications or involve an Unreviewed Safety Question. The following are summaries of significant Plant Procedure (PPM) changes that were processed during 1994.

2.6.6.1
Procedure Revision Form for PPM 1.4.13

This procedure provides instructions for the preparation, control, and revision of Motor Operated Valve (MOV) Master Data Sheets. The revision to this PPM was for the purpose of refining the method of setting the close bypass switch.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this procedure change. The proposed changes to the method of setting the close bypass switch increase the reliability of the valves to operate by ensuring that full motor capability is available to close the valve to flow cut off. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.2
Procedure Revision Forms for PPM's 2.3.3A and 2.3.3B

These procedures provide detailed instructions for Containment Atmosphere Control (CAC) system operation. The revisions to these procedures were for the purpose of administrative and programmatic enhancements. A process of pressurizing the CAC skid to facilitate visual examination of reworked mechanical connections was provided and the process of setting scrubber water flow rate was changed.

It was concluded from the safety evaluations that an unreviewed safety question is not introduced as a result of these procedure changes. The CAC System functions as an accident mitigation system used to assure containment integrity when hydrogen and oxygen gases are generated following a postulated loss-of-coolant accident. The CAC System is normally isolated from the primary containment during Plant Operational Modes 1, 2, or 3. The CAC System does not initiate any design basis event. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.3
Procedure Revision Form for PPM 2.4.6

This procedure provides instructions for the operation of the Reactor Core Isolation Cooling System (RCIC) in all modes of operation. This procedure change incorporated an alternative method for warmup and pressurization of the RCIC steam supply line following an RCIC isolation.

It was concluded from the safety evaluations that an unreviewed safety question is not introduced as a result of this procedure change. Although the alternative method deviates from the original method described in the FSAR, it ensures the original intent of minimizing the possibility of thermal shock or water hammer to the RCIC steam supply piping. The implementing activity affects only the administrative method in which the RCIC system is placed into standby lineup and does not affect the design function or capability of the RCIC system. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.4
Procedure Revision Form for PPMs 4.10.2.5

This procedure provides for the control of the Main Control Room Temperature. This procedure change provides a list of control room loads which may be shed to lower ambient control room temperatures under accident conditions.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The lighting and equipment to be turned off are not safety-related nor do they interact with safety-related systems. In addition, it was concluded that this activity would not impact the consequences of an accident evaluated previously in the LBDs.

2.6.6.5
Procedure Revision Form for PPM 4.601.A4

This procedure provides instructions for the reponse to annunciators on P601. This procedure change provides the direction to measure leak rate from RHR-V-8 and RHR-V-9 and prohibits operation of the RHR system in normal SDC mode after an accident without positive confirmation that RHR-V-165 is closed.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The activity is within the limits established by the failure of the RHR shutdown cooling event which is already analyzed. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.6
Procedure Revision Form for PPM 6.3.34

This is a new procedure developed to support installation of vessel bottom head plug to facilitate maintenance on the vessel drain line.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The plug used was designed and supplied specifically for this application by General Electric. The plug is held in place by gravity and the interference fit of the Buna seal rings. No physical structural attachments are required. The plug itself is purely mechanical and does not require motive force in order to maintain its sealing properties. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.7
Procedure Revision Form for PPM 6.5.18

This is a new procedure developed to delineate activities required to support the vendor jet pump hold down beam replacement during the 1994 refuelling outage.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. The jet pumps do not perform a safety function when the reactor is in mode 5. Technical Specifications do not require them to be operable in mode 5. In addition, it was concluded that this activity would not impact the consequences of an accident evaluated previously in the LBDs.

2.6.6.8
Procedure Revision Form for PPM 7.4.0.5.25

This procedure is for the purpose of controlling reactor pressure/temperature within confines of the Pressure/Temperature Limit curves in the Technical Specifications. This procedure was revised to incorporate the option of using RPV beltline temperature to control vessel temperatures.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. This procedure change does not modify the method of performing the procedure, it only modifies the controls of the procedure. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.9
Procedure Revision Forms for PPMs 7.4.3.1.3.1 through 7.4.3.1.3.4

These procedures provide for the surveillance of APRM Fixed Neutron Flux Upscale Trip Channels. Procedure changes were incorporated to revise the input test voltage for APRM channels A - F for the 118% APRM Fixed Neutron Flux Function from an input signal of 0 - 120% to an input signal of 95 - 125%. Response time, which is listed in Technical Specifications Table 3.3.1-2, will not change.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Since the APRM Fixed Neutron Flux - High function only provides input to the RPS system and does not input or interface with reactivity controls, there is no mechanism which could increase the probability of occurrence of an accident evaluated previously in the LBDs. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.10
Procedure Revision Form for PPM 7.4.4.6.1.1C

This procedure is used to control reactor pressure/temperature within the confines of the Pressure/Temperature Limit curves in Technical Specifications. This procedure revision incorporates the option of using RPV beltline temperature if available to control vessel temperatures during hydrostatic tests.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Three controlling components for the vessel are the flange, the feedwater nozzle and the beltline. For the performance of the non-nuclear heating hydro/leak test, the area above the beltline can be at a lower temperature limited by the next limiting component which would be the feedwater nozzles. The curve limits would still be utilized for the beltline regions of the vessel. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.11
Procedure Revision Form for PPM 8.3.125

This procedure provides instructions for confirming mechanical/electrical overspeed trip capabilities of RCIC-SS-1 and RCIC-SS-2 in their as-is and/or post-maintenance conditions. The purpose of this revision was to reflect the setpoint changes recommended in with General Electric Service Information Letter (SIL) SIL Number 382.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Implementation of this procedure change does not affect the ability of the electronic overspeed trip to perform its design safety function. Therefore, the activity would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.6.12
<u>Procedure Revision Form for PPM 8.3.318</u>

This procedure provides instructions for MOV DP Testing of SSW system valves. This procedure revision directs performance of differential pressure testing of SW-V-187A and SW-V-188A.

It was concluded from the safety evaluation that an unreviewed safety question is not introduced as a result of this change. Performance of differential pressure testing of valves SW-V-187A and SW-V-188A does not require exceeding any operating or system design limits. This test does not increase the probability or consequences of an accident evaluated previously in the LBDs.

2.6.7 Miscellaneous

This section covers other plant activities for which a 10CFR50.59 Safety Evaluation was performed during 1994.

2.6.7.1
Computer Change Request TE-94-015

This computer change request provided for deactivation of the PRIME-based Emergency Dose Projection System (EDPS). The PC-based EDPS function was declared operational and the PRIME version was no longer required.

It was concluded from the safety evaluation that the activity would not increase the probability of occurrence of an accident evaluated previously in the LBDs. Dose projection methodology and related software are informational tools used to formulate protective action recommendations during accidents that involve releases of radioactivity. Therefore, implementation of this activity would not create an unreviewed safety question.

2.6.7.2
ASME Code Relief Request 2ISI-05

This relief request provided an evaluation for taking exception to the 1989 ASME Code pertaining to IWD-5223(f) pneumatic testing following main steam safety relief valve replacement. The test adds unnecessarily to the plant radiological exposure burden (approximately 1.6 person rem per outage).

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs. No system design feature is affected, nor is any mode or method of conducting a plant design safety function influenced by waiver of the pneumatic test.

2.6.7.3
ASME Code Relief Request 2ISI-06

This relief request provided an evaluation for taking exception to the 1989 ASME Code pertaining to the removal of all bolting for examination from the Control Rod Drives when a leak is discovered during Inservice Inspection.

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs because the postulated event is bounded by the limting accident already analyzed in the LBDs. In the event that the eight drive bolts fail, the drive is limited to three inches of travel by the control rod housing support steel which is much more restrictive than a full rod drop accident from a reactivity view point and is already bounded by the LOCA analysis.

2.6.7.4
ASME Code Relief Request 2ISI-07

This relief request provided an evaluation for taking exception to Section XI of the 1989 ASME Code pertaining to the removal of all bolting for examination from the bolted joint when a leak is found during Inservice Inspection.

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs. The activity has the potential to affect small-break LOCAs inside primary containment, however, the changes do not alter the consequences because large-break LOCAs are bounding events. In addition, high energy line breaks in secondary containment have been previously analyzed and are still bounding.

2.6.7.5
ASME Code Relief Request 2ISI-12

This relief request provided an evaluation for taking exception to the 1989 ASME Code pertaining to piping system/component defect removal.

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs. Taking exception to the through-wall examination requirements does not change any of the previous ASME Code and Code Case requirements to which the plant was built and has been maintained. The only accident that could be postulated would be a failure of a pressure boundary weld. However, repair activities would be performed in accordance with all original construction and maintenance requirements.

2.6.7.6
ASME Code Relief Request 2ISI-13

This relief request provided an evaluation for taking exception to the 1989 ASME Code pertaining to alternate pressure testing requirements for welded repairs or replacement of Code items by welding.

It was concluded from the safety evaluation that the activity would not increase the consequences of an accident evaluated previously in the LBDs. Since weld quality is not reduced under the alternative rules of Code Case N-416-1, the probability of pressure boundary weld failure is not increased and, therefore, both the probability of weld failure occurrence and the consequences of such failures are not increased.

2.6.7.7
ASME Code Relief Request 2ISI-14

This relief request provided an evaluation for taking exception to Section XI of the 1989 ASME Code pertaining to having to remove all bolting from a bolted connection when a leak is discovered after repair or replacement.

It was concluded from the safety evaluation that the activity would not increase the probability of an accident previously evaluated in the LBDs. The limiting pressure boundary failure due to bolted connection failure would be an instantaneous, double-ended pipe break or, in this case, a bolted joint separation. Since bolt quality is not reduced under the proposed alternate approach, the probability of a joint failure is not increased.

2.6.7.8
Licensee Controlled Specification 1.7.2.1

The purpose of this activity was to incorporate the operability requirements for the Control Room Emergency Chillers into the Licensee Controlled Specifications.

It was concluded from the safety evaluation that the activity would not increase the probability of an accident evaluated previously in the LBDs. The chillers are used to enhance the working environment of the Control Room during post-accident conditions. Therefore, the probability of an accident is not increased.

2.6.7.9
COLR 94-10

This safety evaluation allowed for implementation of the WNP-2, Cycle 10, Core Operating Limits Report (COLR).

It was concluded from the safety evaluation that the activity would not increase the probability of an accident evaluated previously in the LBDs. Operation of Cycle 10 within the thermal limits defined in COLR 94-10 does not increase the consequences of the analyzed anticipated operational occurrences or accidents because the mechanical, thermal hydraulic and LOCA design criteria imposed on the fuel to protect it during these events are met.

2.6.7.10

Design Specification for Division 60

This safety evaluation allowed for justification for revision to Division 60, "Reactor Core and System Analysis Parameters for WNP-2," to reflect the current values of the parameters used in the transient analysis.

It was concluded from the safety evaluation that the activity would not increase the probability of an accident evaluated previously in the LBDs. The effect of the change to conditions on transient analysis is considered to be small, however, the effect is accounted for in determining the MCPR limits. Furthermore, operation of Cycle 10 within the thermal limits defined in COLR 94-10 does not increase the consequences of the analyzed anticipated operational occurrences or accidents because the mechanical, thermal hydraulic and LOCA design criteria imposed on the fuel to protect it during these events are met.

2.7 REPORT OF DIESEL GENERATOR FAILURES

This section of the report contains information regarding diesel generator failures, valid and non-valid, in accordance with the requirements of WNP-2 Technical Specification 4.8.1.1.3. WNP-2 experienced two non-valid load demand failures in 1994 for the three emergency diesel generators.

2.7.1

- Identity of Diesel Generator and date of failure:

  Division Two Emergency Diesel Generator (DG-2); June 21, 1994 (1126 Hours).

- Number designation of failure in last 100 valid tests:

  This was the first failure of the last 100 tests. However, this test was determined to be a "non-valid" load demand failure.

- Cause of failure:

  During the performance of the annual LOOP/LOCA surveillance test, the Division Two Emergency Diesel Generator failed to take on load, and tripped on reverse power. During troubleshooting efforts, it was determined that the cause was due to a control switch being in the "Isochronous" position, rather than in the "Parallel" position.

- Corrective measures taken:

  The control switch was placed into the "Parallel" position.

- Length of time the Diesel Generator unit was unavailable:

  The Diesel Generator was out of service for approximately one hour and was returned to testing at 1230 hours on June 21, 1994.

- Current surveillance test interval:

  Thirty-one days.

2.7.2

- Identity of Diesel Generator and date of failure:

  Division Two Emergency Diesel Generator (DG-2); June 21, 1994 (1308 Hours).

- Number designation of failure in last 100 valid tests:

  This was the second failure of the last 100 tests. However, this test was also determined to be a "non-valid" load demand failure.

- Cause of failure:

  During the performance of the annual LOOP/LOCA surveillance test, the Division Two Emergency Diesel Generator failed to take on load, and tripped on reverse power. The troubleshooting efforts revealed that the Metering and Relaying Potential Transformers (PTs) were indicating a loss of one phase. It was determined that the cause was due to a misalignment of the secondary stabs on the PTs, which simulated a reverse power condition at the protective relay.

- Corrective measures taken:

  The PT stabs were cleaned and realigned to ensure adequate engagement with the fixed contact blocks.

- Length of time the Diesel Generator unit was unavailable:

  The Diesel Generator was out of service for approximately 97 hours and was returned for testing at 1358 hours on June 25, 1994.

- Current surveillance test interval:

  Thirty-one days.