WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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March 1, 1994
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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 1993

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 1993. Should you have any questions or desire additional information regarding this matter, please contact Mr. John D. Arbuckle at (509) 377-4601.

Sincerely,

Gregory O Smith/For

J.V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

Enclosure 070133

cc: KE Perkins, NRC - Region V
REIRs Project Manager, NRC - NRR
RF Mazurkiewicz, BPA (MD 399)
DL Williams, BPA (MD 399)
NRC Site Inspector (MD 927N)

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WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1993

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 1993 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.

The plant ran at, or near, 100 percent power until January 21, 1993 when a forced outage occurred due to the inadvertent initiation of the Fire Protection Deluge System in a Reactor Feedwater (RFW) pump room. The initiation caused the loss of one RFW Turbine which, in turn, resulted in a low reactor water level scram from 100 percent power. Following the necessary repairs, the plant was returned to service on January 29, 1993. During February 1993, the plant experienced two additional forced shutdowns caused by) a manual scram when a Reactor Recirculation pump failed to tran er to 60 Hz operation due to an interlock signal in the pump control circuitry, and 2) an automatic scram on low level due to the loss of an RFW pump which was caused by the failure of an electrical coil in the RFW pump governor control circuitry. Following the completion of an assessment, the plant resumed full power operation during mid-February 1993.

On April 30, 1993 the plant was shutdown for the annual maintenance and refueling outage. In the remaining months of the year following the outage, the plant experienced one forced outage on August 3, 1993 due to an automatic scram which was caused by a full isolation of the Main Steam Isolation Valves (MSIVs) during the performance of a surveillance procedure to calibrate a main steam radiation indicating switch. During troubleshooting efforts, it was determined that an MSIV pilot valve had been improperly rebuilt during the recent refueling outage. Following repairs, the plant was restarted and resumed full power operation by month's end. During the remainder of the year, the plant ran at, or near, 100 percent power and new monthly records were set for electrical generation and plant capacity factor.

During 1993 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 eighth refueling outage was successfully completed and significant activities included:

- A complete overhaul of the Reactor Feedwater Turbine.

- Cleaning of five Reactor Recirculation System Jet Pumps.

- Draining and cleaning of one of the two Standby Service Water System Spray Ponds.

- Improvements to switchyard insulator design.

- Replacement of 128 fuel assemblies.

In addition, the plant achieved a 75 percent capacity factor and received industry-wide recognition of its excellent thermal efficiency. A summary of thermal performance prepared by the Institute of Nuclear Power Operations (INPO) listed WNP-2 as the best in the United States in this category for boiling water reactors.

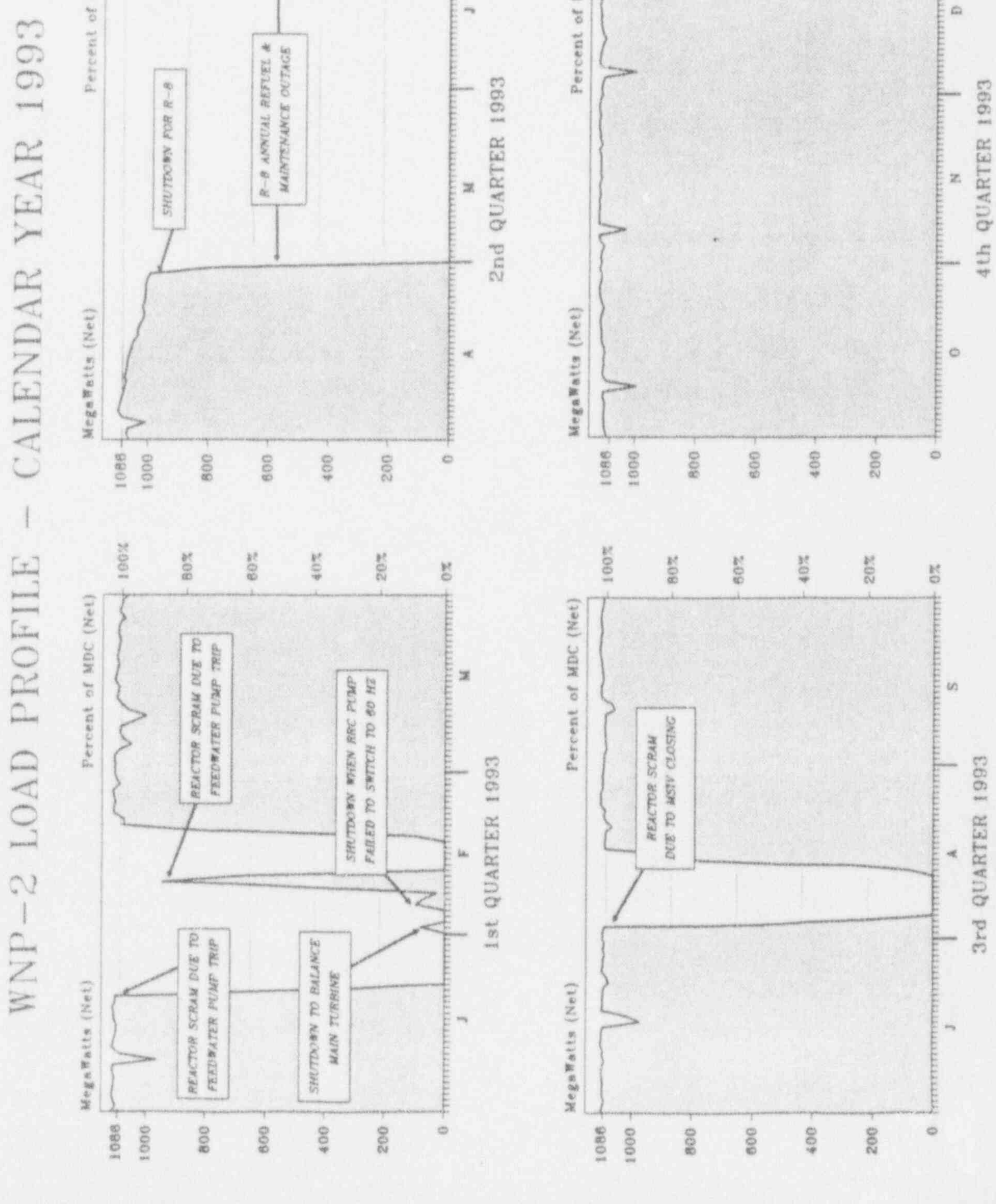
The 1993 capacity factors, based on net electrical energy output are listed below.

Month	Capacity Factor
January	64.66
February	45.37
March	97.77
April*	93.50
May	0
June**	17.09
July	100.28
August	64.87
September	100.90
October	101.67
November	102.54
December	102.34
Overall	75.00

* Started Maintenance and Refueling Outage
** Ended Maintenance and Refueling Outage

NOTE: Capacity factors for 1993 were based on a Maximum Dependable Capacity (MDC) of 1086 MWe.

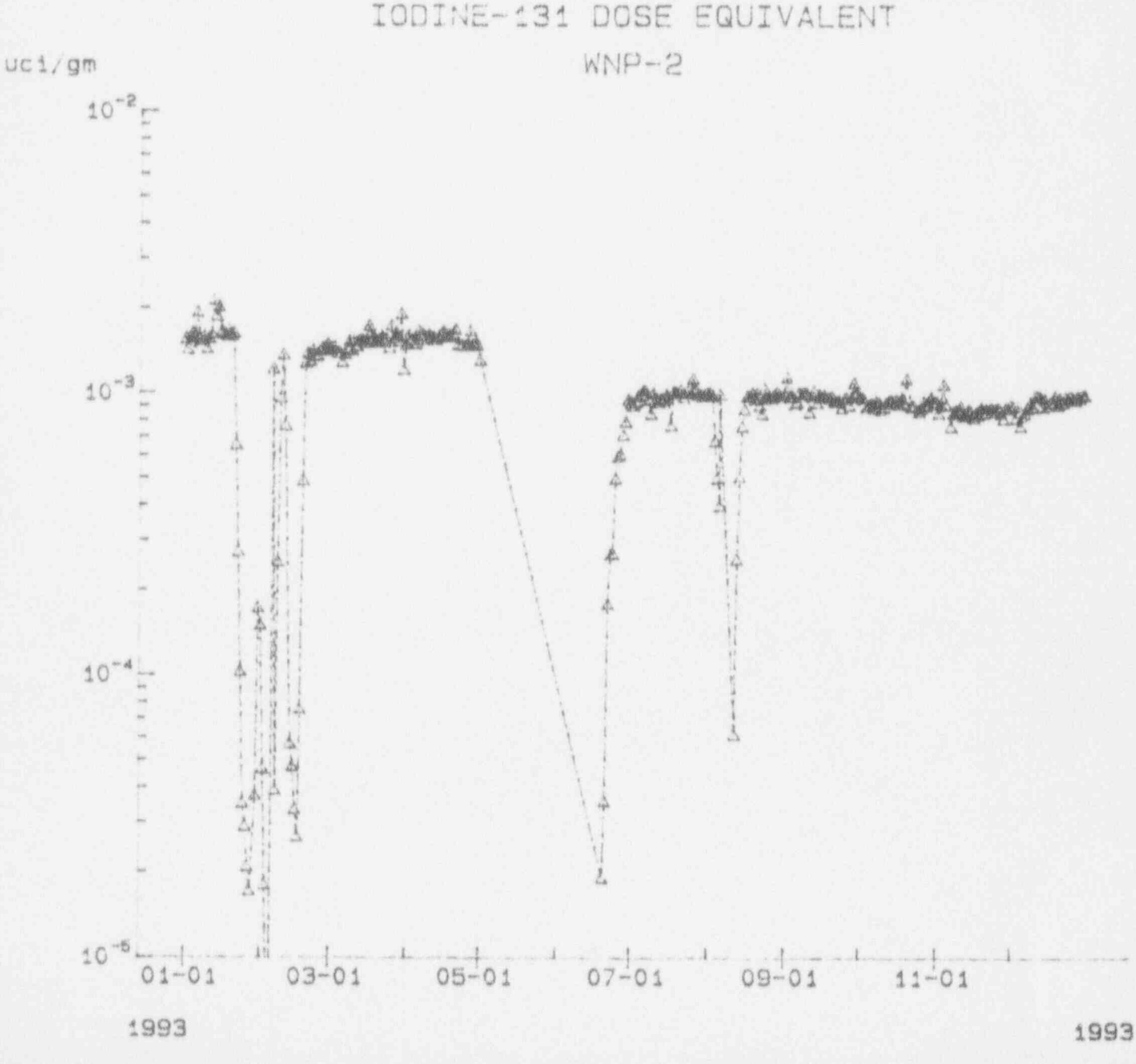
1.1 WNP-2 LOAD PROFILE FOR 1993



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.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT / 1.16 APPENDIX A

NUCLEAR PLANT NO. 2 — REPORT FOR CALENDAR YEAR 1993

		NUMBER OF PERSONS RECEIVING OVER 100 MREM			TOTAL MAN-REM		
		STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS
OPERATIONS & SURVEILLANCE	MAINTENANCE PERSONNEL	0.281	0.000	0.036	0.178	0.000	0.021
	OPERATING PERSONNEL	0.320	0.000	0.000	0.252	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.497	0.000	0.189	0.346	0.000	0.178
	SUPERVISORY PERSONNEL	0.604	0.000	0.000	0.144	0.000	0.000
	ENGINEERING PERSONNEL	0.014	0.108	0.000	0.004	0.081	0.000
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	248.995	6.616	260.874	153.751	3.060	136.977
	OPERATING PERSONNEL	59.169	6.046	0.022	33.959	1.156	0.004
	HEALTH PHYSICS PERSONNEL	45.179	1.667	51.156	27.107	0.344	29.419
	SUPERVISORY PERSONNEL	18.938	2.457	1.370	5.138	0.582	0.395
	ENGINEERING PERSONNEL	17.185	26.598	45.531	4.746	8.644	18.057
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	0.009	0.000	13.373	0.008	0.000	5.170
	OPERATING PERSONNEL	0.012	0.000	0.000	0.008	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.124	0.000	0.384	0.127	0.000	0.144
	SUPERVISORY PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
	ENGINEERING PERSONNEL	0.519	1.633	4.651	0.246	0.697	3.081
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	5.723	0.240	1.080	3.862	0.098	0.595
	OPERATING PERSONNEL	0.068	0.000	0.000	0.042	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.753	0.000	0.728	0.627	0.000	0.314
	SUPERVISORY PERSONNEL	1.040	0.000	0.000	0.285	0.000	0.000
	ENGINEERING PERSONNEL	1.507	0.553	0.694	0.612	0.166	0.136
WASTE PROCESSING	MAINTENANCE PERSONNEL	6.330	0.070	0.000	3.426	0.013	0.000
	OPERATING PERSONNEL	0.019	0.000	0.000	0.013	0.000	0.000
	HEALTH PHYSICS PERSONNEL	1.921	0.000	1.020	1.045	0.000	2.182
	SUPERVISORY PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
	ENGINEERING PERSONNEL	0.046	0.043	0.014	0.021	0.013	0.008
REFUELING	MAINTENANCE PERSONNEL	2.789	0.074	12.058	1.416	0.017	2.715
	OPERATING PERSONNEL	2.626	0.040	0.000	2.157	0.004	0.000
	HEALTH PHYSICS PERSONNEL	0.207	0.000	2.871	0.090	0.000	0.909
	SUPERVISORY PERSONNEL	0.593	0.000	0.630	0.217	0.000	0.072
	ENGINEERING PERSONNEL	0.815	2.133	8.758	0.123	0.574	1.455
TOTAL	MAINTENANCE PERSONNEL	264.127	7.000	287.421	162.641	3.188	145.478
	OPERATING PERSONNEL	62.214	6.086	0.022	36.431	1.160	0.004
	HEALTH PHYSICS PERSONNEL	48.681	1.667	56.348	29.342	0.344	33.146
	SUPERVISORY PERSONNEL	21.175	2.457	2.000	5.784	0.582	0.467
	ENGINEERING PERSONNEL	20.086	31.068	59.648	5.752	10.175	22.737
GRAND TOTAL		416.283	48.278	405.439	239.950	15.449	201.832

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 1993 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	63790-00-0059	63790-00-0058	63790-00-0048		
Component ID (Location)	MS-RV-4A	MS-RV-4C	MS-RV-1A		
Date of Actuation (MO/DA/YR)	02/10/93	02/10/93	02/10/93		
Time of Day (24 Hour Clock)	1832	1849	1922		
Type of Actuation (Code)	B	B	B		
Cause/Reason for Actuation (Code)	E	E	E		
Rx Operating Condition Prior to Lift (Code)	G	G	G		
Rx Power Level Prior to Lift (% Rated Thermal)	0 (Decay Heat)	0 (Decay Heat)	0 (Decay Heat)		
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A		
Other Instrumentation Type (Code)	PROCESS COMPUTER	PROCESS COMPUTER	PROCESS COMPUTER		
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN		
Rx Pressure Prior to Actuation (PSIG)	605	610	600		
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A		
Duration of This Actuation (Minutes, Seconds)	3 min, 39 sec	4 min, 52 sec	3 min, 13 sec		
Failures, Reports (Code)	C	C	C		
LER Number (5 Digit Number)	None	None	None		
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes		

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

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0061	63790-00-0126	63790-00-0062	63790-00-0060	63790-00-0056
Component ID (Location)	MS-RV-5B	MS-RV-3D	MS-RV-5C	MS-RV-4D	MS-RV-4B
Date of Actuation (MO/DA/YR)	02/10/93	02/10/93	02/10/93	02/10/93	02/10/93
Time of Day (24 Hour Clock)	1750	1751	1754	1805	1816
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	E	E	E	E	E
Rx Operating Condition Prior to Lift (Code)	G	G	G	G	G
Rx Power Level Prior to Lift (% Rated Thermal)	0 (Decay Heat)	0 (Decay Heat)	0 (Decay Heat)	0 (Decay Heat)	0 (Decay Heat)
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)	900	840	620	595	600
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 min, 10 sec	4 min, 11 sec	1 min, 29 sec	3 min, 32 sec	4 min, 1 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0049	63790-00-0053	63790-00-0061	63790-00-0046	63790-00-0051
Component ID (Location)	MS-RV-2B	MS-RV-3B	MS-RV-5B	MS-RV-1C	MS-RV-3C
Date of Actuation (Mo/Da/Yr)	5/1/93	5/1/93	5/1/93	5/1/93	5/1/93
Time of Day (24 Hour Clock)	0909	0915	0918	0920	0957
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)	D	D	D	D	D
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)	A	A	A	A	A
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	918	919	918	919	918
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	4 sec	4 sec	4 sec
Failures, Reports (Code)	C	C	C	B	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

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	63790-00-0054	63790-00-0055
Component ID (Location)	MS-RV-5C	MS-RV-4D	MS-RV-1A	MS-RV-2A	MS-RV-3A
Date of Actuation (Mo/Da/Yr)	5/1/93	5/1/93	5/1/93	5/1/93	5/1/93
Time of Day (24 Hour Clock)	0957	0957	0957	1027	1027
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)	D	D	D	D	D
Rx Power Level Prior to Lift (% Rated Thermal)	$\approx$15%	$\approx$15%	$\approx$15%	$\approx$15%	$\approx$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)	A	A	A	A	A
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	918	918	919	918	918
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	4 sec	4 sec	4 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0059	63790-00-0045	63790-00-0056	63790-00-0047	63790-00-0058
Component ID (Location)	MS-RV-4A	MS-RV-1B	MS-RV-4B	MS-RV-2C	MS-RV-4C
Date of Actuation (Mo/Da/Yr)	5/1/93	5/1/93	5/1/93	5/1/93	5/1/93
Time of Day (24 Hour Clock)	1027	1027	1027	1027	1027
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)	D	D	D	D	D
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)	A	A	A	A	A
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	918	918	918	918	918
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	4 sec	4 sec	4 sec
Failures, Reports (Code)	C	B	C	C	B
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0050	63790-00-0126	63790-00-0124	63790-00-0048	63790-00-0054
Component ID (Location)	MS-RV-1D	MS-RV-3D	MS-RV-2D	MS-RV-1A	MS-RV-2A
Date of Actuation (Mo/Da/Yr)	5/1/93	5/1/93	5/1/93	6/20/93	6/20/93
Time of Day (24 Hour Clock)	1027	1027	1027	1927	1927
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)	D	D	D	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	≈15%	≈15%	≈15%	≈10%	≈10%
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	A	A	A	A	A
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	918	918	918	920	920
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	4 sec	8 sec	4 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0049	63790-00-0053	63790-00-0061	63790-00-0046	63790-00-0058
Component ID (Location)	MS-RV-2B	MS-RV-3B	MS-RV-5B	MS-RV-1C	MS-RV-4C
Date of Actuation (Mo/Da/Yr)	6/20/93	6/20/93	6/20/93	6/20/93	6/20/93
Time of Day (24 Hour Clock)	1927	1927	1927	1927	1927
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)	≈10%	≈10%	≈10%	≈10%	≈10%
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	A	A	A	A	A
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	920	920	920	920	920
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	4 sec	6 sec	6 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	. 3	4	5
S/R Valve Serial Number	63790-00-0062	63790-00-0124	63790-00-0126		
Component ID (Location)	MS-RV-5C	MS-RV-2D	MS-RV-3D		
Date of Actuation (Mo/Da/Yr)	6/20/93	6/20/93	6/20/93		
Time of Day (24 Hour Clock)	1927	1927	1927		
Type of Actuation (Code)	C	C	C		
Cause/Reason for Actuation (Code)	C	C	C		
Rx Operating Condition Prior to Lift (Code)	C	C	C		
Rx Power Level Prior to Lift (% Rated Thermal)	≈10%	≈10%	≈10%		
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A		
Other Instrumentation Type (Code)	A	A	A		
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN		
Rx Pressure Prior to Actuation (PSIG)	920	920	920		
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A		
Duration of This Actuation (Minutes, Seconds)	12 sec	4 sec	4 sec		
Failures, Reports (Code)	B	C	C		
LER Number (5 DIgit Number)	None	None	None		
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes		

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0053	63790-00-0060	63790-00-0062	63790-00-0059	63790-00-0052
Component ID (Location)	MS-RV-3B	MS-RV-4D	MS-RV-5C	MS-RV-4A	MS-RV-3C
Date of Actuation (Mo/Da/Yr)	6/21/93	6/21/93	6/21/93	6/21/93	6/. ./93
Time of Day (24 Hour Clock)	0920	0927	0936	0942	1025
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)	950	950	950	950	950
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)	34 sec	1 min, 45 sec	56 sec	1 min, 55 sec	24 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0048	63790-00-0046	63790-00-0049	63790-00-0126	63790-00-0056
Component ID (Location)	MS-RV-1A	MS-RV-1C	MS-RV-2B	MS-RV-3D	MS-RV-4B
Date of Actuation (Mo/Da/Yr)	6/21/93	6/21/93	6/21/93	6/21/93	6/21/93
Time of Day (24 Hour Clock)	1028	1032	1036	1039	1045
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)	950	950	950	950	950
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)	18 sec	23 sec	18 sec	46 sec	1 min, 13 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0057	63790-00-0058	63790-00-0050	63790-00-0061	63790-00-0124
Component ID (Location)	MS-RV-3A	MS-RV-4C	MS-RV-1D	MS-RV-5B	MS-RV-2D
Date of Actuation (Mo/Da/Yr)	6/21/93	6/21/93	6/21/93	6/21/93	6/21/93
Time of Day (24 Hour Clock)	1048	1052	1056	1059	1104
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)	950	950	950	950	950
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)	14 sec	1 min. 10 sec	16 sec	50 sec	18 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	None	None	None	None	None
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0120	63790-00-0122	63790-00-0054		
Component ID (Location)	MS-RV-1B	MS-RV-2C	MS-RV-2A		
Date of Actuation (Mo/Da/Yr)	6/21/93	6/21/93	6/21/93		
Time of Day (24 Hour Clock)	1107	1110	1114		
Type of Actuation (Code)	B	B	B		
Cause/Reason for Actuation (Code)	C	C	C		
Rx Operating Condition Prior to Lift (Code)	C	C	C		
Rx Power Level Prior to Lift (% Rated Thermal)	≈15%	≈15%	≈15%		
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A		
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer		
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN		
Rx Pressure Prior to Actuation (PSIG)	950	950	950		
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A		
Duration of This Actuation (Minutes, Seconds)	16 sec	27 sec	38 sec		
Failures, Reports (Code)	C	C	C		
LER Number (5 Digit Number)	None	None	None		
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes		

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0046	63790-00-0050	63790-00-0049	63790-00-0122	63790-00-0048
Component ID (Location)	MS-RV-1C	MS-RV-1D	MS-RV-2B	MS-RV-2C	MS-RV-1A
Date of Actuation (Mo/Da/Yr)	8/03/93	8/03/93	8/03/93	8/03/93	8/03/93
Time of Day (24 Hour Clock)	0539	0539	0539	0539	0539
Type of Actuation (Code)	A	A	A	A	A
Cause/Reason for Actuation (Code)	A	A	A	A	A
Rx Operating Condition Prior to Lift (Code)	E	E	E	E	E
Rx Power Level Prior to Lift (% Rated Thermal)	100%	100%	100%	100%	100%
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	CLOSED
Rx Pressure Prior to Actuation (PSIG)	1083	1089	1089	1089	1089
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	1002	1020	1020	1020	N/A
Duration of This Actuation (Minutes, Seconds)	7 Sec	7 Sec	6 Sec	6 Sec	0
Failures, Reports (Code)	C	C	C	C	A
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0120	63790-00-0048	63790-00-0126	63790-00-0059	63790-00-0056
Component ID (Location)	MS-RV-1B	MS-RV-1A	MS-RV-3D	MS-RV-4A	MS-RV-4B
Date of Actuation (Mo/Da/Yr)	8/03/93	8/03/93	8/03/93	8/03/93	8/03/93
Time of Day (24 Hour Clock)	0539	0652	0539	0601	0551
Type of Actuation (Code)	A	B	B	B	B
Cause/Reason for Actuation (Code)	A	E	E	E	E
Rx Operating Condition Prior to Lift (Code)	E	E	E	E	E
Rx Power Level Prior to Lift (% Rated Thermal)	100%	Decay Heat	Decay Heat	Decay Heat	Decay Heat
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	CLOSED	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	1089	1009	1060	989	1017
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	937	884	824	826
Duration of This Actuation (Minutes, Seconds)	0	53 Sec	37 Sec	2 Min 17 Sec	3 Min 21 Sec
Failures, Reports (Code)	A	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0058	63790-00-0061	63790-00-0062	63790-00-0120	
Component ID (Location)	MS-RV-4C	MS-RV-5B	MS-RV-5C	MS-RV-1B	
Date of Actuation (Mo/Da/Yr)	8/03/93	8/03/93	8/03/93	8/06/93	
Time of Day (24 Hour Clock)	0616	0539	0539	1840	
Type of Actuation (Code)	B	B	B	B	
Cause/Reason for Actuation (Code)	E	E	E	D	
Rx Operating Condition Prior to Lift (Code)	E	E	E	G	
Rx Power Level Prior to Lift (% Rated Thermal)	Decay Heat	Decay Heat	Decay Heat	0	
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	
Instrumentation ber Reading and Units	OPEN	OPEN	OPEN	OPEN	
Rx Pressure Prior to Actuation (PSIG)	1008	1074	1044	0	
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	826	916	815	N/A	
Duration of This Actuation (Minutes, Seconds)	3 Min 56 Sec	25 Sec	2 Min 11 Sec	≈ 30 Sec	
Failures, Reports (Code)	C	C	C	C	
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	

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

CODES:

Type of Actuation

A. Automatic
B. Remote Manual
C. Spring

Plant Condition

A. Construction
B. Startup or Power Ascension Tests in Progress
C. Routine Startup
D. Routine Shutdown
E. Steady State Operation
F. Load Changes During Routine Operation
G. Shutdown (Hot or Cold)
H. Refueling

Reason for Actuation

A. Overpressure
B. ADS or Other Safety System
C. Test
D. Inadvertent (Accidental/Spurious)
E. Manual Relief

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: JANUARY 1993

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
93-01	1/21/93	F	199.71	H	3	93-002	CH	PUMPXX	INADVERTENT INITIATION OF FIRE PROTECTION DELUGE SYSTEM IN FEEDPUMP ROOM CAUSED A LOSS OF FEEDWATER PUMP, RESULTING IN LOW REACTOR WATER LEVEL SCRAM FROM 100% POWER. AFTER REPAIR/DRYOUT OF FEEDPUMP CIRCUITRY AND RESOLUTION OF VARIOUS OTHER MAINTENANCE PROBLEMS, THE PLANT WAS RETURNED TO SERVICE ON 1/29/93.
93-02	1/29/93	S	52.45	B	2	N/A	CC	PIPEXX	REACTOR WAS PLACED IN COLD SHUTDOWN FOR REPAIR OF A STEAM LEAK UPSTREAM OF MS-V-239. AFTER REPAIR OF THE CRACKED WELD, THE PLANT WAS RETURNED TO SERVICE.

SUMMARY: WNP-2 INCURRED ONE FORCED OUTAGE AND ONE SCHEDULED OUTAGE IN IN JANUARY AS DESCRIBED ABOVE.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F-FORCED	A-EQUIP FAILURE	F-ADMIN	1-MANUAL	EXHIBIT F & H
S-SCHED	B-MAINT OR TEST	G-OPER ERROR	2-MANUAL SCRAM	INSTRUCTIONS FOR
	C-REFUELING	H-OTHER	3-AUTO SCRAM	PREPARATION OF
	D-REGULATORY RESTRICTION		4-CONTINUED	DATA ENTRY SHEET
	E-OPERATOR TRAINING & LICENSE EXAM		5-REDUCED LOAD	LICENSEE EVENT REPORT
			9-OTHER	(LER) FILE (NUREG -0161)

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)

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

REPORT PERIOD: **FEBRUARY, 1993**

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
93-03	2/01/93	S	92.74	B	1	N/A	HA	Turbine	Turbine bearing vibration reached unacceptable levels, resulting in plant shutdown. After completion of turbine balance, plant startup was further delayed due to containment isolation valve problems. After valves were repaired, the plant was returned to service.
93-04	2/06/93	F	32.42	A	2	93-006	CB	Pumpxx	Reactor was manually scrammed when reactor recirc pump failed to transfer to 60 HZ operation due to false signal (noise) in the pump control circuitry.
93-05	2/10/93	F	160.38	A	3	93-007	CH	Pumpxx	Reactor automatically scrammed on low level due to loss of reactor feedwater pump. The cause of failure was an electric coil in the feed pump governor control circuitry. After repairs were performed, startup activities were placed on administrative hold while an assessment was performed on recent plant problems. After completion of assessment, the plant was returned to service.

SUMMARY:

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F-FORCED	A-EQUIP FAILURE	F-ADMIN	1-MANUAL	EXHIBIT F & H
S-SCHED	B-MAINT OR TEST	G-OPER ERROR	2-MANUAL SCRAM	INSTRUCTIONS FOR
	C-REFUELING	H-OTHER	3-AUTO SCRAM	PREPARATION OF
	D-REGULATORY RESTRICTION		4-CONTINUED	DATA ENTRY SHEET
	E-OPERATOR TRAINING & LICENSE EXAM		5-REDUCED LOAD	LICENSEE EVENT REPORT
			9-OTHER	(LER) FILE (NUREG -0161)

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)

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

REPORT PERIOD: **MARCH, 1993**

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
NONE									

SUMMARY: WNP-2 operated near full power during March with no outages or significant power reductions.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F-FORCED	A-EQUIP FAILURE	F-ADMIN	1-MANUAL	EXHIBIT F & H
S-SCHED	B-MAINT OR TEST	G-OPER ERROR	2-MANUAL SCRAM	INSTRUCTIONS FOR
	C-REFUELING	H-OTHER	3-AUTO SCRAM	PREPARATION OF
	D-REGULATORY RESTRICTION		4-CONTINUED	DATA ENTRY SHEET
	E-OPERATOR TRAINING & LICENSE EXAM		5-REDUCED LOAD	LICENSEE EVENT REPORT
			9-OTHER	(LER) FILE (NUREG -0161)

2.3 **SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)**

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

EPORT PERIOD: **APRIL, 1993**

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
NONE									

SUMMARY;

WNP-2 operated routinely during April. ˉntered EOC coastdown mode on April 6 and coastdown continued until start of shutdown for refueling outage R-8 on April 30.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
-FORCED	A-EQUIP FAILURE	F-ADMIN	1-MANUAL	EXHIBIT F & H
-SCHED	B-MAINT OR TEST	G-OPER ERROR	2-MANUAL SCRAM	INSTRUCTIONS FOR
	C-REFUELING	H-OTHER	3-AUTO SCRAM	PREPARATION OF
	D-REGULATORY RESTRICTION		4-CONTINUED	DATA ENTRY SHEET
	E-OPERATOR TRAINING & LICENSE EXAM		5-REDUCED LOAD	LICENSEE EVENT REPORT
			9-OTHER	(LER) FILE (NUREG -0161)

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)

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

REPORT PERIOD: **MAY 1993**

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
93-06	5/01/93	S	728.2	C	1	n/a	RC	FUELXX	Plant was shutdown as scheduled for the annual refueling outage R-8.

SUMMARY: WNP-2 operated on May 1, 1993 just a few hours before it was shutdown for the annual refueling outage R-8.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F - Forced	A - Equipment Failure	E - Operator Training & License Examination	1 - Manual	NUREG-0161 Exhibits F & H
S - Scheduled	B - Maintenance or Test	F - Administration	2 - Manual Scram	
	C - Refueling	G - Operational Error	3 - Auto Scram	
	D - Regulatory Restriction	H - Other	4 - Continued	
			5 - Reduced Load	

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: **June 1993**

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
93-06	5/01/93	S	500.7	C	4	n/a	RC	FUELXX	Plant completed annual refueling outage R-8.

SUMMARY: WNP-2 returned to service from the annual refueling outage R-8.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F - Forced	A - Equipment Failure	E - Operator Training & License Examination	1 - Manual	NUREG-0161 Exhibits F & H
S - Scheduled	B - Maintenance or Test	F - Administration	2 - Manual Scram	
	C - Refueling	G - Operational Error	3 - Auto Scram	
	D - Regulatory Restriction	H - Other	4 - Continued	
			5 - Reduced Load	
			9 - Other	

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: **JULY 1993**

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
NONE									

SUMMARY: WNP-2 operated near full power during the month of July with no outages or significant power reductions (>20%).

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F - Forced S - Scheduled	A - Equipment Failure B - Maintenance or Test C - Refueling D - Regulatory Restriction	E - Operator Training & License Examination F - Administration G - Operational Error H - Other	1 - Manual 2 - Manual Scram 3 - Auto Scram 4 - Continued 5 - Reduced Load 9 - Other	NUREG-0161 Exhibits F & H

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: AUGUST 1993

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
93-07	8/03/93	F	219.5	A	3	93-027	CD	VALVEX	During performance of a surveillance procedure to calibrate Main Steam Radiation Indicating Switch, the reactor unexpectedly scrammed from 100% power due to a full isolation of the MSIVs. It was discovered that a MSIV pilot valve had been improperly rebuilt during the recent refueling outage. It was correctly rebuilt and the plant was eventually restarted. See LER 93-027 for other corrective actions taken and training evolutions planned.

SUMMARY: WNP-2 had one automatic scram during the month of August. Upon completion of required equipment repairs, the plant was restarted and ramped up to full power operation by month's end.

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F - Forced S - Scheduled	A - Equipment Failure B - Maintenance or Test C - Refueling D - Regulatory Restriction	E - Operator Training & License Examination F - Administration G - Operational Error H - Other	1 - Manual 2 - Manual Scram 3 - Auto Scram 4 - Continued 5 - Reduced Load	NUREG-0161 Exhibits F & H

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)

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

REPORT PERIOD: SEPTEMBER 1993

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence

NONE

SUMMARY: WNP-2 operated near full power during the month of September with no outages or significant power reductions (>20%).

TYPE	REASON		METHOD	SYSTEM & COMPONENT
F - Forced	A - Equipment Failure	E - Operator Training & License Examination	1 - Manual	NUREG-0161 Exhibits F & H
S - Scheduled	B - Maintenance or Test	F - Administration	2 - Manual Scram	
	C - Refueling	G - Operational Error	3 - Auto Scram	
	D - Regulatory Restriction	H - Other	4 - Continued	
			5 - Reduced Load	

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: **OCTOBER 1993**

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
NONE									

SUMMARY: WNP-2 operated near full power during the month of October with no outages or significant power reductions (>20%).

TYPE
F - Forced
S - Scheduled

REASON
A - Equipment Failure
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training & License Examination
F - Administration
G - Operational Error
H - Other

METHOD
1 - Manual
2 - Manual Scram
3 - Auto Scram
4 - Continued
5 - Reduced Load

SYSTEM & COMPONENT
NUREG-0161 Exhibits F & H

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/ REDUCTIONS)

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

REPORT PERIOD: **NOVEMBER 1993**

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence

NONE

SUMMARY: WNP-2 operated near full power during the month with no outages or significant power reductions (>20%).

TYPE
F - Forced
S - Scheduled

REASON
A - Equipment Failure
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training & License Examination
F - Administration
G - Operational Error
H - Other

METHOD
1 - Manual
2 - Manual Scram
3 - Auto Scram
4 - Continued
5 - Reduced Load
9 - Other

SYSTEM & COMPONENT
NUREG-0161 Exhibits F & H

2.3 SUMMARY OF PLANT OPERATIONS (SHUTDOWNS/REDUCTIONS)

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

REPORT PERIOD: DECEMBER 1993

No.	Date	Type	Hours	Reason	Method	LER Number	System	Component	Cause and Corrective Action To Prevent Recurrence
NONE									

SUMMARY: WNP-2 operated near full power during the month with no outages or significant power reductions (>20%).

TYPE
F - Forced
S - Scheduled

REASON
A - Equipment Failure
B - Maintenance or Test
C - Refueling
D - Regulatory Restriction
E - Operator Training & License Examination
F - Administration
G - Operational Error
H - Other

METHOD
1 - Manual
2 - Manual Scram
3 - Auto Scram
4 - Continued
5 - Reduced Load
9 - Other

SYSTEM & COMPONENT
NUREG-0161 Exhibits F & H

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

This information is provided in accordance with the requirements of Regulatory Guide 1.16, Revision 4, Section C.1.b(2)(e). In addition to safety-related equipment, components considered to be essential for power generation are also included.

Component	Failure Date	Description
APRM-E/S-PS51	02/23/93	During performance of a Technical Specification surveillance to verify operability of the Average and Local Power Range Monitors (APRMs/LPRMs) prior to a plant startup, it was observed that Ion Chamber Power Supply APRM-E/S-PS51 had failed low. The cause of the failure was attributed to normal wear. The ion chamber power supply was replaced and no further problems were identified.
CSP-V-9	06/08/93	During the performance of a local leak rate test on Suppression Chamber Vacuum Relief Valve CSP-V-10, valve CSP-V-9 was discovered to be leaking past the seat. The cause of the failure was attributed to normal seal wear. The seals were replaced, the limit switches were adjusted to provide for proper indication and no further problems were identified.
DLO-PS-3B1	05/07/93	During post-maintenance testing following a diesel-generator overhaul, Low Oil Pressure Switch DLO-PS-3B1 would not reset. During troubleshooting efforts, the switch was found to be out of calibration. The switch was replaced and no further problems were identified.

DLO-P-3B1	05/21/93	During performance of the annual loss of power test on the diesel-generator following maintenance activities, the coupling failed on Diesel Lube Oil Pump DLO-P-3B1. The failure mode was inconclusive. The ccupling was replaced, the pump/motor assembly was aligned and no further problems were identified.
HPCS-42-4A2E	06/11/93	During performance of motor-operated valve testing on the High Pressure Core Spray (HPCS) condensate storage cross-tie valves, Circuit Breaker HPCS-42-4A2E was found to be tripping below the correct overcurrent setpoints. Following inspection, the line-side phase connections were found to be loose. The breaker was replaced and no further problems were identified.
HPCS-42-4A3A	06/11/93	During performance of motor-operated valve testing on the HPCS condensate storage cross-tie valves, Circuit Breaker HPCS-42-4A3A was found to be tripping below the correct overcurrent setpoints. Following inspection, the line-side phase connections were found to be loose. The breaker was replaced and no further problems were identified.
IRM-TA-2D	01/21/93	During performance of a channel functional test on a Intermediate Range Monitoring (IRM) Channel 'D,' the rod-out-block and downscale indication failed. The cause of the failure was attributed to failed relays in IRM Trip Unit IRM-TA-2D. The relays were replaced and the surveillance was successfully completed.
RCC-M-P/1A	02/22/93	During a plant tour, an equipment operator noted that

		Reactor Closed Cooling (RCC) Water Pump Motor RCC-M-P/1A was exhibiting excessive noise. During troubleshooting efforts, it was discovered that a motor bearing showed signs of lubrication breakdown. The inboard and outboard motor bearings were replaced.
RCC-M-P/1A	03/24/93	During a plant tour, an equipment operator noted that RCC-M-P/1A was again exhibiting excessive noise. During troubleshooting efforts it was determined that the problem was due to under-loading of the newly-installed bearings. The motor bearings were replaced with Rollway bearings and no further problems were identified.
RHR-MO-6A	05/31/93	During motor-operated valve testing, it was noted that a contact on the limit switch rotor for Residual Heat Removal (RHR) System Motor Operator RHR-MO-6A was not closing when the associated valve was fully closed. During troubleshooting efforts, it was discovered that the stationary contacts were out of alignment and not making proper contact. The stationary contacts were reinstalled and aligned, and no further problems were identified.
RHR-MO-73B	05/17/93	Plant Operations personnel reported a loss of control power for RHR Heat Exchanger Shell-Side Vent Valve Motor RHR-MO-73B. During troubleshooting efforts, a wire was discovered to be pinched under the limit switch cover. The pinched wire was repaired and the valve was returned to service.

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 8 was performed in the month of January 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 8 were inspected. No evidence of rod bow, abnormal fuel rod growth, mechanical damage or offset tie rod latches were noted during the inspection of the assemblies.

The fuel channels inspected displayed a uniform covering of light oxidation on unwelded surfaces. However, the heat-effected zone of the weld surface was clean and consistent with past inspections. In addition, there was no observable mechanical damage to the channels.

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 1993 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) or Request For Technical Services (RFTS). The following PMRs/BDCs/RFTSs implemented in 1993 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 87-0244-OE

This BDC provided for the installation of the Reactor Recirculation (RRC) System Adjustable Speed Drive (ASD) building, associated new transformers and fire protection system.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident previously analyzed in the Licensing Basis Documents (LBDs). The modifications are housed in a non-safety related system that is not housed in a safety-related structure.

2.6.1.2
BDC 91-0100-OA

This BDC provided for the replacement of Containment Atmosphere Control (CAC) System ASME, Section VIII relief valves, with ASME, Section III, Code Class 2, relief valves.

It was concluded from the safety evaluation that the replacement of

Valves CAC-V-65A and CAC-V-65B were seismically evaluated and the proposed activity would not increase the consequences of an accident previously evaluated in the LBDs.

2.6.1.3
BDC 90-0305-OA

This BDC provided for the addition of a Gamma Spectroscopy Monitoring System to measure Reactor Building effluents.

It was concluded from the safety evaluation that the installation would not increase the consequences of an accident. The system as installed by this design change has no active function and is only used for monitoring purposes.

2.6.1.4
BDC 55-2006-OA

This BDC provided for the incorporation of the results, into Certified Vendor Information (CVI) drawings, of a calculation pertaining to Limitorque motor-operator housing cover bolt torque values.

It was concluded from the safety evaluation that no physical change 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 motor-operated valve failure and, as a result, it would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.5
BDC 92-0214-0A

This BDC provides 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 refueling accident are independent of the refueling mast design. Therefore, this design change will not impact the consequences of an accident.

2.6.1.6
BDC 92-0297-OA

This BDC provided for replacement of the existing radiators on Transformer TR-N2 with larger radiators.

It was concluded from the safety evaluation that the design function of the transformer is not changed and there are no interactions with safety-related systems or equipment. Therefore, the probability of occurrence of an accident would not be changed.

2.6.1.7
BDC 93-0082-0

This BDC provided for modification of the Reactor Core Isolation Cooling (RCIC) System design such that containment integrity would not be compromised by a single failure. The modification consists of installation of a two-inch, motor-operated isolation valve and local pressure indicator in the cooling water supply line to the RCIC Turbine Auxiliaries.

It was concluded from the safety evaluation that neither of the new components increases the probability of an accident, because the RCIC System is credited as accident mitigation for the Anticipated Transient Without Scram (ATWS) and remote shutdown scenarios.

2.6.1.8
BDC 90-0307-0A

This BDC provided for the installation of a 12-point terminal strip to facilitate calibration of Containment Monitoring System (CMS) instruments CMS-MV/I-41AR and CMS-MV/I-44AR.

It was concluded from the safety evaluation that the addition of a terminal strip in the Suppression Pool temperature monitoring circuits would not increase the probability of occurrence of an accident. This is a non-safety related circuit and the consequences of an event would be decreased by implementation of the design change.

2.6.1.9
BDC 90-0305-3D

This BDC provided for installation of the Main Control Room portion of the Stack Monitoring System. The installation consisted of the mounting of a computer with monitor, modules, power supply and flow recorder in the Control Room.

It was concluded from the safety evaluation that these activities would not increase the consequences of an accident previously evaluated, and no other safety-related components or structures would be affected by the implementing activity.

2.6.1.10
BDC 90-0305-5F

This BDC provided for installation of the low-range detector for Stack Monitor PRM-RE-1A. Existing detectors CMS-RE-27C and CMS-RE-27D were removed and the new Low Range Detector PRM-RE-1A was installed.

It was concluded from the safety evaluation that the modification would not affect core reactivity, the ability to monitor core power or any of the automatic reactivity control functions. Therefore, the probability of accidents previously evaluated was unchanged and not increased.

2.6.1.11
BDC 93-0052-OA

This BDC provided for the use of torque switch control for closing Valve HPCS-V-4 instead of the position limit switch. Use of torque switch control for closing the valve is the preferred control method because full seating of the valve is assured.

It was concluded from the safety evaluation that closure of the valve by the torque switch instead of the limit switch would not increase the probability of occurrence of an accident. Furthermore, the valve closure function remained unchanged by the modification.

2.6.1.12
BDC 55-2427-OA

This BDC provided for revision of top-tier drawings to depict the electronic drain traps in the Control and Service Air (CAS) System dryers.

It was concluded from the safety evaluation that there are no accidents associated with the CAS System and the system would not be lost if one of the electronic drain traps on the refrigerated dryers fails to function properly. Therefore, this design update would not increase the probability of an accident.

2.6.1.13
BDC 55-2804-OA

This BDC provided for revision of drawings to incorporate calculated bolt torque recommendations for motor-operator valves subjected to 140 percent of operator-rated thrust.

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-operated valve bolts, which are supported by calculation.

2.6.1.14
BDC 87-0242-OB

This BDC provided for installation of four low-pressure sodium security light fixtures adjacent to the new ASD Building.

It was concluded from the safety evaluation that the design change has no direct or indirect interface with structures, systems or components for which the failure would initiate an accident evaluated previously. The change would not alter the consequences of any previously evaluated design basis accident.

2.6.1.15
BDC 92-0300-OA

This BDC provided for replacement of the existing Main Steam Line Radiation Monitor with a NUMAC Log Radiation Monitor LRM.

It was concluded from the safety evaluation that the interface with the main steam ion detectors and associated trip functions would not be changed. The monitor would perform in accordance with the existing accident analysis and the consequences of an accident previously evaluated would not be increased.

2.6.1.16
BDC 93-0089-OB/OC

This BDC provides for the installation of a continuous backfill modification which will flow Control Rod Drive (CRD) System water into five water level instrument reference legs.

It was concluded from the safety evaluation that there are no new bypass leakage paths cause by installing this modification. In addition, all previous accident analyses are still bounding and the activity would not prevent any equipment important to safety from performing the intended safety function.

2.6.1.17
BDC 91-0143-OA

This BDC provided for independent adjustment of Main Steam (MS) System Motor-Operator MS-MO-146 bypass and position indicating limit switch trip settings.

It was concluded from the safety evaluation that the design change has no affect on the intended safety function of Valve MS-V-146.

Therefore, the previously-evaluated dose limit as described in the LBDs would remain unchanged.

2.6.1.18
BDC 91-0212-OA

This BDC provided for relocation of valve control and indication functions for Residual Heat Removal (RHR) Valves RHR-V-21 and RHR-V-48A.

It was concluded from the safety evaluation that the design change has no affect on the intended safety function of the valves to mitigate a LOCA. The change results in enhanced operability and a more accurate indication of valve position.

2.6.1.19
BDC 92-0216-OA

This BDC provided for replacement of the existing limit switch for Valve RCIC-V-1 position indication with a qualified unit.

It was concluded from the safety evaluation that the qualified switch provides more reliable control and decreases the probability of occurrence of malfunction of equipment.

2.6.1.20
BDC 90-0018-OA

This BDC provided for installation of Local Area Network (LAN) components inside the plant for the protected area to have access to various application computer programs.

It was concluded from the safety evaluation that the LAN is part of the Telecommunication System and it does not perform any safety-related function and would not affect any structures, systems or components important to safety.

2.6.1.21
BDC 87-0048-3D

This BDC provided for the revision of the design database and replacement of components in the Supervisory Control System (Division A). The Supervisory Control System was replaced with an new system, that performs the same function, to provide for increased reliability through a modular arrangement and state-of-the-art design.

It was concluded from the safety evaluation that the proposed activity would not increase the probability of an accident evaluated previously in the LBDs. The effect of this activity serves to improve the reliability of the non-safety related Supervisory Control System.

2.6.1.22
BDC 91-0125-OA

This BDC provided for the installation of a permanent shield support structures around Reactor Recirculation (RRC) System piping inside primary containment.

It was concluded from the safety evaluation that installation of the support structures would not increase the probability of occurrence of an accident previously evaluated in the LBDs. A target walkdown and safe shutdown analysis for impacts on the support structures from postulated pipe breaks and jet impingement was completed. The ability to safely shut down the plant should one of these postulated events occur was demonstrated through this analysis.

2.6.1.23
BDC 91-0221-OA

This BDC provided for the addition of an interlock to the automatic-close circuits for Standby Service Water (SSW) System Valves SW-V-12A and SW-V-12B. The interlock will prevent the valves from "hammering" on valve closure.

It was concluded from the safety evaluation that the modification would increase nuclear safety because it will decrease "hammering" of the valves on the closing cycle. This activity will decrease component wear and increase circuit reliability. Furthermore, it was concluded that the modification would not compromise the ability of the SSW System to supply water to those systems under any accident evaluated previously in the LBDs.

2.6.1.24
BDC 92-0086-OA

This BDC provided for the replacement of Relief Valve RHR-RV-36 with a blind flange structural assembly (RHR-TPSA-1). The relief valve, which served no containment isolation function, was located in the deactivated RHR Steam Condensing Mode piping.

It was concluded from the safety evaluation that this modification would not increase the probability of occurrence of an accident previously evaluated in the LBDs. The blind flange structural assembly would perform the same function as RHR-RV-36.

2.6.1.25
<u>BDC 92-0189-OA</u>

This BDC provided for proper electrical separation of three Reactor Closed Cooling (RCC) and four Reactor Building HVAC System circuit breakers by adding series isolation fuses and the rewiring of the control and trip circuits such that separation of redundant systems would be achieved.

It was concluded from the safety evaluation the proposed activity would provide for conformance to the electrical separation requirements between redundant divisions and;therefore, would not increase the probability of occurrence of an accident previously evaluated in the LBDs.

2.6.1.26
<u>BDC 92-0238-OA/OB</u>

This BDC provided for the replacement of trim and actuator parts in RFW Heater Level Control Valves so that the valves will be suitable for operation under uprated plant conditions.

It was concluded from the safety evaluation that the type of valves, the valve operators and the functions of the valves would remain unchanged. Therefore, the probability of occurrence of an accident evaluated previously in the LBDs would not be increased by the proposed activity.

2.6.1.27
<u>BDC 93-0024-OA</u>

This BDC provided for the revision of the Diesel Generator (DG) start logic by providing additional safety-related logic to terminate field flash, coincident with operation of the voltage regulator.

It was concluded from the safety evaluation that the addition of logic relays to terminate the field flash interval ensures that DG-2 would start and supply voltage to the accident mitigation equipment within the accident-specified time limits.

2.6.1.28
<u>BDC 93-0113-OA/OB</u>

This BDC provided for installation of low-point drain lines on the Containment Atmosphere Control (CAC) System hydrogen recombiners to remove any accumulation of condensate in the low-point piping prior to recycling through the blower and/or returning to containment.

It was concluded from the safety evaluation that the addition of the drain lines would ensure that the recombiners are capable of performing the post-LOCA function of ensuring that a combustible mixture is not created in containment.

2.6.1.29
BDC 93-0117-OA

This BDC provided for the redesign of the instrument tap connections for RHR Differential Indicating Switch RHR-DPIS-12B.

It was concluded from the safety evaluation that the change would increase the strength and reliability of the sensing lines. Therefore, the modification would not increase the consequences of any accident previously evaluated in the LBDs.

2.6.1.30
BDC 93-0141-OA

This BDC provided for increase of the torque output of High Pressure Core Spray (HPCS) System Motor-Operator HPCS-MO-12 to assure for proper valve operation under design basis conditions.

It was concluded from the safety evaluation that the increase in stroke time from four to eight seconds would not affect the response time of the HPCS System to an initiation signal. The eight-second stroke time is well within the 20-second maximum time for containment isolation. Therefore, the consequences of any accidents requiring initiation of HPCS would remain unchanged.

2.6.1.31
PMR 89-0100-0

This PMR provided for installation of a level indicator for Glycol Tank GY-TK-2.

It was concluded from the safety evaluation that implementation of the design change would not result in degradation of nuclear safety as described in the LBDs. The addition of the glycol storage tank level gauge would not cause any increase in the consequences of an accident evaluated previously in the LBDs.

2.6.1.32
PMR 89-0151-1

This PMR provided for the addition of drain lines connecting to the Main Condenser return lines at two locations to reduce the accumulation of condensate in the turbine and seal steam pressure sensing lines.

It was concluded from the safety evaluation that implementation of the design change would not result in degradation of nuclear safety as described in the LBDs. The drain lines will improve the reliability of the RFW Turbine by reducing erosion caused by condensate accumulation.

2.6.1.33
PMR 92-0120-1

This PMR provided for the change-out of fuses for motor-operated valves that were to be refurbished during the 1993 refueling outage.

It was concluded from the safety evaluation that implementation of the design change would not result in degradation of nuclear safety as described in the LBDs. Reducing the fuse size would enhance the protection of the motors against a locked-rotor or locked-armature current condition.

2.6.1.34
PMR 93-0066-0

This PMR provided for change-out of the gears in RHR-MO-16A, RHR-MO-16B, RHR-MO-17A and RHR-MO-17B to increase the torque and thrust output.

It was concluded from the safety evaluation that implementation of the design change would not result in degradation of nuclear safety as described in the LBDs. Implementation of this modification would result in improved performance and reliability of the RHR containment spray loop.

2.6.1.35
PMR 93-0061-0

This PMR provided for modification of the air removal off-take lines in, and adjacent to, the Main Condenser.

It was concluded from the safety evaluation that implementation of the design would not result in degradation of nuclear safety as described in the LBDs. The line configuration would be slightly changed to permit greater air removal capacity from the lower banks of the condenser.

2.6.1.36
PMR 90-0134-0

This PMR provided for installation of permanent thermowells and resistance temperature detectors in the Service Water and Diesel Cooling Water (DCW) lines to the DCW heat exchangers.

It was concluded from the safety evaluation that implementation of the design would not result in degradation of nuclear safety as described in the LBDs. The purpose of the change is to provide for the means to obtain data for performance monitoring evaluations of the DCW heat exchangers.

2.6.1.37
PMR 88-0038-34

This PMR provided for installation of new microprocessor-based recorders at the Meteorological Tower.

It was concluded from the safety evaluation that this primarily non-safety related modification would not increase the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.1.38
PMR 92-0159-OB

This PMR provided for replacement of 14 existing 1.5 hour battery-powered emergency lighting units with eight-hour units.

It was concluded from the safety evaluation that the modification cannot increase the probability of an accident since the lighting is only for accident mitigation purposes.

2.6.1.39
PMR 92-0222-0

This PMR provided for taking credit for the use of existing Thermo-Lag plugs in the end of rigid conduit runs for secondary containment conduit seals to allow the rework of fire seals for the installation of watertight seals.

It was concluded from the safety evaluation that the rework of the existing seals would increase the margin of safety associated with the ECCS pump room flooding issue. Therefore, once installed, the watertight seals would decrease the consequences of an accident.

2.6.1.40
PMR 89-0151-1

This PMR provided for installation of drain connections from RFW-DT-1A and RFW-DT-1B low points to the Main Condenser return lines.

It was concluded from the safety evaluation that the modification would reduce the probability of the occurrence of an RFW turbine trip by reducing turbine damage.

2.6.1.41
PMR 93-0158-0A

This PMR provided for replacement of the step-down and regulating transformers for Power Panel E-PP-7BC with a single 30 kva transformer and re-powering of the Stack Monitoring System.

It was concluded from the safety evaluation that the new identified loads were evaluated for voltage drop, cable ampacity, panel ratings, transformer ratings and diesel generator loading and all values were within acceptable limits.

2.6.1.42
PMR 90-0361-OA

This PMR provided for changing several motor-operated Service Water System valves to manually operated valves.

It was concluded from the safety evaluation that changing the valves from electrically operated to manual operation (and locked in the open position) would not change the safety function of the valves, nor would it decrease the ability of the valves to perform their intended safety function.

2.6.1.43
PMR 91-0231-OA

This PMR provided for the addition of a tee in the tubing run to the pilot valve to allow for in-service checks and calibration of pressure control valves in the Diesel Starting Air (DSA) System.

It was concluded from the safety evaluation that the consequences of an accident would not be increased since the change does not affect the Diesel Generator units and; therefore, does not affect the ability of the units to support a LOCA/LOOP scenario.

2.6.1.44
PMR 90-0018-O

This PMR provided for the removal of three secondary containment penetration air and fire seals to facilitate installation of new communication cables in the Reactor Building.

It was concluded from the safety evaluation that the planned removal of the seals would not create an increase in the probability of occurrence of an accident. Furthermore, an hourly fire tour would be in place during removal and replacement efforts.

2.6.1.45
PMR 87-0326-0
This PMR provided for the replacement of the Diesel Generator combustion air oil bath filters with cartridge-type filters.

It was concluded from the safety evaluation that changing the type of filter would not interfere with the ability of the Diesel Generator units to carry the required emergency electrical loads.

2.6.1.46
PMR 92-0184-OA

This PMR provided for the replacement of three-position control switches with a maintained contact two-position switch configuration for the Diesel Fuel Oil Day Tank Room Fans.

It was concluded from the safety evaluation that the incorporation of the proposed design would serve to ensure that the day tank room exhaust fans are operable following a LOOP. Therefore, there would be no increase of the consequences of an accident previously evaluated in the LBDs.

2.6.1.47
PMR 88-0038-53

This PMR provided for the replacement of the Turbine Building stack exhaust pneumatic-controlled flow monitor with an electronic recorder.

It was concluded from the safety evaluation that the proposed activity is designed to standards consistent with current design requirements and, therefore, would not increase the consequences of an accident evaluated previously in the LBDs.

2.6.1.48
RFTS 92-11-008

This RFTS provided for installation of temporary shielding in the RRC System to shield a hotspot, resulting in minimization of radiation exposure to station personnel.

It was concluded from the safety evaluation that the shielding would be installed and secured such that it would not contribute to the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.1.49
RFTS 92-11-011

This RFTS provided for installation of temporary shielding of reactor well drain piping to minimize radiation exposure to station personnel.

It was concluded from the safety evaluation the shielding would be installed and secured such that it would not contribute to the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.1.50
RFTS 93-01-119

This RFTS provided for installation of temporary shielding of an equipment drain line to minimize radiation exposure to station personnel.

It was concluded from the safety evaluation that the shielding would be installed and secured such that it would not contribute to the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.2 TEMPORARY MODIFICATIONS AND SETPOINT CHANGES

The following are summaries of temporary modifications and 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 and certain setpoint changes are made under the Motor-Operated Valve Setpoint Change Request (MSCR) process.

2.6.2.1
TMR 93-028

This TMR provided for the connection of a transmitter to the high-point vent located at Control Rod Drive (CRD) System Valve CRD-V-522, for the purposes of data collection pertaining to the Reactor Pressure Vessel (RPV) water level instrument reference leg backfill system.

It was concluded from the safety evaluation that this activity would not increase the consequences of an accident previously evaluated and provides input to the overall effort to install a back-fill modification.

2.6.2.2
TMR 93-025

This TMR provided for the installation of a flow control station associated with tubing and supports for the Reactor Pressure Vessel (RPV) level instrumentation modification. The RPV level instrumentation reference leg purge system design was limited to the 522′ elevation in the Reactor Building.

It was concluded from the safety evaluation that the modification would not affect core reactivity, the ability to monitor core power or any of the automatic reactivity control functions. Therefore, the probability of accidents previously evaluated was unchanged and not increased.

2.6.2.3
TMR 93-30

This TMR provided for the installation of test leads to spare normally-closed contacts on the open and close limit switches of Scram Discharge Volume (SDV) vent and drain valves to support performance of stoke-time testing.

It was concluded from the safety evaluation that the activity would not increase the probability of occurrence of an accident, and that installation of the temporary test loads on the electrically-isolated limit switches of the valves provide a reliable means to perform surveillance testing without exposing plant personnel to high levels of radiation.

2.6.2.4
MSCR 370

This MSCR provided for changing the thrust setpoint for Auxiliary Steam (AS) System Motor Operator AS-MO-68A following a change to thrust setpoint methodology and reviews of system design bases.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident and the MSCR provides additional margin for valve operation at design basis conditions.

2.6.2.5
MSCR 371

This MSCR provided for changing the thrust setpoint for Motor Operator AS-MO-68B following a change to thrust setpoint methodology and reviews of system design bases.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident of a different type and the MSCR provides additional margin for valve operation at design basis conditions.

2.6.2.6
MSCR 380

This MSCR provided for changing the torque switch setting for Service Water (SW) System Motor Operator SW-MO-69B.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident, and the recommended setpoint provides increased assurance that the valve will function on demand.

2.6.2.7
MSCR 381

This MSCR provided for changing the torque switch setting for Motor Operator SW-MO-70B.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident and the recommended setpoint provides increase assurance that the valve would function on demand.

2.6.3 FSAR CHANGES

General Changes to the FSAR evaluated within the definition of 10CFR50.59 are reported in this section.

2.6.3.1
SCN 93-001

This SCN provided for revision of the Diesel Generator loading schedules due to revised calculations.

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.

2.6.3.2
SCN 93-033

This SCN provided for clarification of Control Room temperature information as it relates to emergency cooling equipment.

It was concluded from the safety evaluation that the temperatures are within the Technical Specification limits. Furthermore, the revision clarifies the FSAR to accurately define the temperatures which the Emergency Cooling systems are capable of maintaining in the Control Room.

2.6.3.3
SCN 93-034

This SCN provided for clarification of the design description pertaining to the acceptability of 32 degree fahrenheit water flowing through the tube side of the Residual Heat Removal (RHR) System Heat Exchanger.

It was concluded from the safety evaluation that impact test results on the materials of the heat exchanger displayed a large margin and that the 32-degree value was acceptable. Therefore, under accident conditions, the heat exchanger would be able to perform its safety function.

2.6.3.4
SCN 93-038

This SCN provided for the revision of the discussion of the Containment Instrument Air (CIA) System backup nitrogen bottles pertaining to capacity and capability, as opposed to the number of bottles.

It was concluded from the safety evaluation that the commitment to provide for a 30-day supply of nitrogen was not being altered. The description was being modified to avoid confusion and to merely allow for provisions for one bottle at a time to be valved out-of-service for maintenance activities.

2.6.3.5
SCN 93-039

This SCN provided for removal of the remaining references to chlorine detection in the Control Room remote air intake system.

It was concluded from the safety evaluation that the need for the chlorine detectors was eliminated by a modification which replaced the gaseous chlorination system, for the Tower Service Water System, with a liquid chlorination system. Accordingly, this activity would not increase the consequences or probability of an accident previously evaluated in the Licensing Basis Documents (LBDs).

2.6.3.6
SCN 93-037

This SCN provided for the authorization for the application of Ameron Amercoat 90 over power-tool cleaned steel inside the primary containment.

It was concluded from the safety evaluation that the proposed change is qualified the same as the original coating; therefore, there was no increase in the consequences of any previously-evaluated accident.

2.6.3.7
SCN 93-044

This SCN provided for changing of close-time settings for RHR Motor Operators RHR-MO-16A, RHR-MO-16B, RHR-MO-17A and RHR-MO-17B from ten seconds to the manufacturer's standard speed setting of 12 inches per minute for gate valves.

It was concluded that the proposed activity would not increase the consequences of an accident evaluated previously in the LBDs. This activity would reduce the possibility of failure of the motor operators by modifying the gear sets and spring pack so that the valves are more likely to be able to open under the operating thrust and torque requirements.

2.6.3.8
SCN 93-049

This SCN provided for deletion of the requirement to close the Main Steam System Trap Station valves.

It was concluded from the safety evaluation that the proposed activity would not change the intent or commitments described in the LBDs, and the margin of safety would not be reduced.

2.6.3.9
SCN 93-058

This SCN provided for revision of site hazards, Regulatory Guide positions, Anticipated Transient Without Scram (ATWS) analysis and Safety Evaluation Report (SER) statements.

It was concluded from the safety evaluation the proposed activity would not increase the probability of occurrence of an accident or increase the consequences of design basis accidents.

2.6.3.10
SCN 93-060

This SCN provided for revision of the main plant release point Effluent Monitor System description.

It was concluded from the safety evaluation that the proposed activity would not increase the probability of occurrence of an accident or increase the consequences of design basis accidents. The new monitoring system will provide a more immediate indication of the release of effluents through the Reactor Building Elevated Release Duct.

2.6.3.11
SCN 93-068

This SCN provided for revision to show that the manual valves in the Service Water-to-Fuel Pool Cooling makeup water supply line are normally closed, and need to be opened following a LOCA to allow for remote-manual makeup capability to the spent fuel pool.

It was concluded from the safety evaluation that the proposed activity was within previously analyzed limits and, as a result, there would be no increase in the consequences of an accident previously evaluated.

2.6.3.12
SCN 93-97

This SCN provided for changing Reactor Core Isolation Cooling (RCIC) System Valve RCIC-V-8 close time from ten seconds to 16 seconds.

It was concluded from the safety evaluation that, even with the increased stroke time, all equipment remained qualified to perform its required safety function as defined in the LBDs. Therefore, the consequences of an accident previously evaluated will not change.

2.6.3.13
SCN 93-101

This SCN provided for clarification of testing requirements of post-accident valves.

It was concluded from the safety evaluation that removal of the periodic channel check and channel calibration testing requirements for position indication would not increase the probability of an accident.

2.6.3.14
SCN 93-113

This SCN provided for elimination of the piping connection from a Reactor Water Cleanup (RWCU) System line to Flow Control Valve RWCU-FCV-33 from consideration as a postulated terminal-end High Energy Line Break (HELB) location.

It was concluded from the safety evaluation that the implementing activity eliminates a potential accident location from consideration and;therefore, would not increase the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.3.15
COLR 93-9

This safety evaluation allowed for implementation of the WNP-2, Cycle 9, Core Operating Limits Report (COLR).

It was concluded from the safety evaluation that the proposed activity would not increase the probability of occurrence of previously evaluated accidents. The thermal limits for the Cycle 9 reload core were developed with an NRC-approved methodology and would 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 have been met to protect it during any such events.

2.6.4 PROBLEM EVALUATIONS

The Plant Problems-Plant Problem Reports Procedures (PPMs 1.3.12A and 1.3.12B) 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 292-1406

This PER documented a situation where it was discovered as part of a walkdown that field-installed fuses, in Class 2 Distribution Panel E-DP-S1/2b, were not of the type and size as specified in the design documents. The safety evaluation allowed for the fuses to remain installed until they could be replaced with the preferred fuses during the next outage of sufficient duration.

It was concluded from the safety evaluation that faults or failures on these Class 2 branch circuits would not impact cabling servicing equipment important to safety and, as a result, would not increase the consequences of an accident.

2.6.4.2 PER 292-1443

This PER was written to evaluate the adequacy of the 20 micron cartridge filter in the suction piping of the Post Accident Sampling System (PASS) sample pumps for the five Reactor Building sumps and the Suppression Pool.

It was concluded from the safety evaluation that the presence or absence of filters in the sump samples could not have any effect on the probability of occurrence of an accident. The sample lines which included the filters are not in use during normal operation, with the possible exception of a few minutes every three months.

2.6.4.3 PER 292-1444

This PER was written to request an operability assessment of safety-related, motor-operated valves with SMB/SB-000 and SMB/SB/SBD-00 Limitorque actuators during a seismic/hydrodynamic event.

It was concluded from the safety evaluation that none of the 91 safety-related valves identified are an initiator for anticipated operational transients or postulated design basis accidents.

Therefore, an increase in valve opening or closing time during a seismic/hydrodynamic event would not affect the probability of an accident.

2.6.4.4
PER 293-0044

This PER documented a situation where it was discovered as part of a walkdown that field-installed fuses, in Distribution Panel E-DP-S1/2A, were not of the type and size as specified in the design documents. The safety evaluation allowed for the fuses to remain installed until they could be replaced with the preferred fuses during the next outage of sufficient duration.

It was concluded from the safety evaluation that faults or failures of non-Class 1E branch circuits would not impact the operation of equipment important to safety and, as a result, would not increase the consequences of an accident.

2.6.4.5
PER 293-0180

This PER documented a situation where it was discovered that the long-time trip setting on the solid state trip device for six branch Motor Control Center (MCC) feeder breakers was slightly out-of-calibration high. The safety evaluation allowed for continued operation.

It was concluded from the safety evaluation that the failure of these breakers to coordinate on a fault could not create an accident, and there would be no increase in the previously evaluated consequences of an accident.

2.6.4.6
PER 293-0183

This PER was written to request revision of the setpoint and allowable value for Reactor Core Isolation Cooling (RCIC) System Pressure Switches RCIC-PS-22A, RCIC-PS-22B, RCIC-PS-22C, and RCIC-PS-22D to 75 psig and 60 psig respectively.

It was concluded from the safety evaluation that revising the trip setpoints to a value such that the trip will occur slightly earlier could not create an accident. Furthermore, existing accident analyses and consequences were unchanged by these setpoint changes.

2.6.4.7
PER 293-0279

This PER documented a situation where it was discovered as part of a walkdown that field-installed fuses, in Power Panel E-PP-8-AA, were not of the type and size as specified in the design documents. The safety evaluation allowed for the fuses to remain installed until they could be replaced with the preferred fuses during the next outage of sufficient duration.

It was concluded from the safety evaluation that adequate coordination existed such that failure of non-Class 1E equipment would not impact Class 1E circuits. As a result, allowing the fuses to remain installed would not increase the consequences of malfunction of equipment important to safety.

2.6.4.8
PER 293-0299

This PER documented a situation where a chemistry sample confirmed that the glycol inside of the Agitator Off-Gas Refrigeration System was contaminated.

It was concluded from the safety evaluation that the probability of the occurrence of an accident would not be increased due to the contamination.

2.6.4.9
PER 293-0301

This PER documented a situation where it was discovered that reduced power MCPR limits at 25 percent power in the Core Operating Limit Report (COLR) were based on an incorrect analysis.

It was concluded from the safety evaluation that the proposed activity, of adding a conservative penalty to the MCPR Operating Limit, would not increase the probability of occurrence of an accident as addressed in the LBDs.

2.6.4.10
PER 293-0366

This PER documented a situation where it was determined that flexible conduit installed for certain instrumentation exceeded the maximum allowable span as dictated by procedure. The safety evaluation allowed for the installation of tie-wraps on the flexible conduits at the instrument racks.

It was concluded from the safety evaluation that the probability of the occurrence of an accident would not increase the probability of

2.6.4.11
PER 293-0498

This PER documented a situation where it was discovered that the setpoint for High Pressure Core Spray (HPCS) System Relay HPCS-RLY-TD3/DG3 was set outside of the relay range. The setpoint was subsequently changed.

It was concluded from the safety evaluation that the setpoint change still allowed for three, one-second intervals for the pinion to engage in the flywheel. Therefore, the consequences of an accident evaluated previously in the LBDs would not be increased.

2.6.4.12
PER 293-0604

This PER documented a situation where it was discovered that the replacement hinge arm for Residual Heat Removal (RHR) System Valve RHR-V-89 did not properly fit into the valve. The disc position ring, linkage and indicating rod were removed from the valve.

It was concluded from the safety evaluation that the removal of the "testable" position indication features of the valve would not increase the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.4.13
PER 293-0678

This PER documented a situation where it was determined that, in response to NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," the existing analyses did not address the consequences on strainer head loss due to the filtering of sludge and other debris. The safety evaluation allowed for the operability of the ECCS Systems until the nest refueling outage.

It was concluded from the safety evaluation that allowing plant operation would not increase the probability of occurrence of an accident evaluated previously in the LBDs. The systems and components affected provide mitigation functions for accidents. There are no credible mechanisms resulting from the existence of sludge in the wetwell pool which could increase the probability of an accident.

2.6.4.14
PER 293-0711

This PER documented a situation where it was determined that a top plate of a Pipe Whip Support (PWS) in the Reactor Feedwater (RFW) System should be eliminated.

It was concluded from the safety evaluation that removal of the plate would not increase the probability of occurrence of an accident evaluated previously in the LBDs, nor would it affect the consequences of accidents previously evaluated in the LBDs.

2.6.4.15
PER 293-0710

This PER was written to consider the removal of a removable concrete block wall in the Reactor Building to facilitate snubber inspections.

It was concluded from the safety evaluation that the proposed activity would not increase the consequences of an accident previously evaluated in the LBDs. Removal of the wall would not affect the radiological consequences, or reduce the ability to mitigate radiological consequences, of any accident. Furthermore, the wall is not a primary or secondary containment barrier.

2.6.4.16
PER 293-0747

This PER documented a situation where, during post-modification testing, Valve SW-V-12A failed to automatically close completely. The safety evaluation allowed for declaration of the operable status of SW System (Loop A) with the valve tagged full open.

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. Service Water, Loop A, operability would be unaffected by the potential of SW-V-12A not going to the full-close position, or by deenergizing the valve in the full-open position.

2.6.4.17
PER 293-0866

This PER was written to allow for permanent modification of the temporarily-disabled Refueling Bridge Load Float circuit. Disabling of the load float circuitry was recommended by a General Electric Service Information Letter (GE SIL 503).

It was concluded from the safety evaluation that the proposed activity would not increase the probability of occurrence of an accident previously evaluated in the LBDs. Having the load switch in an operable status could result in the bypassing of several refueling interlocks which would, in turn, increase the probability of accidents previously evaluated.

2.6.4.18
PER 293-0901

This PER was written to document the use of Class 2 gears in a Class 1 motor operator on HPCS-V-12. The safety evaluation provided the basis for continued operation.

It was concluded from the safety evaluation that the proposed activity would not increase the consequences of an accident evaluated previously in the LBDs. The motor-operator would be expected to function as designed using the Class 2 gears. Furthermore, engineering calculations determined that the gears could withstand an adequate number of cycles for the motor-operator to perform its safety function.

2.6.4.19
PER 293-0909

This PER documented a situation where, during the performance of leakage inspections in the steam tunnel, Main Steam (MS) System Valve MS-V-20 exhibited packing gland leakage. The safety evaluation allowed for the on-line repair of the valve by means of Furmanite until permanent repairs could be made.

It was concluded from the safety evaluation that the proposed repair to the valve packing chamber would not affect the pressure retaining boundary or increase the probability of a valve rupture.

2.6.4.20
PER 293-0924

This PER documented a situation where it was determined that five solid state relays in the Containment Monitoring System (CMS) could energize and activate the H2/O2 sample trouble alarm due to normal output current leakage. The safety evaluation allowed for the addition of a drainage circuit around the alarm relays.

It was concluded from the safety evaluation that the devices would be seismically mounted and electrically isolated by a Class 1E fuse. Accordingly, there would be no change in the probability of a malfunction of other equipment in the system.

2.6.4.21
PER 293-1026

This PER documented a situation where it was discovered that, due to a change in the construction of certain overload relays manufactured after 1980, the overload heater sizing tables used by WNP-2 were incorrect. The safety evaluation allowed for the installation of newer model heaters.

It was concluded from the safety evaluation that installing heater sizes in the new model would provide the same operability margins and protection levels as the original models. Accordingly, there would be no change in the probability of occurrence of an accident.

2.6.4.22
PER 293-1046

This PER documented a situation where it was noted that the support steel for a hanger had pulled away from the wall, resulting in damage to concrete and the snubber. The safety evaluation allowed for deletion of the snubber.

It was concluded from the safety evaluation that there was no increase in the probability of occurrence of an accident evaluated previously in the LBDs because system configuration continued to meet the structural requirements of applicable codes and standards.

2.6.4.23
PER 293-1050

This PER documented a situation where a small steam leak was observed in a weld area upstream of Valve MS-V-22A. The safety evaluation allowed for repair of the small-bore instrument line.

It was concluded from the safety evaluation that the repair of the instrument line would return the system to a state equivalent to its original configuration. Therefore, the proposed activity would not create the possibility of a different type of malfunction of equipment important to safety than any evaluated previously in the LBDs.

2.6.4.24
PER 293-1176

This PER documented a situation where it was discovered that a 480 volt feeder cable was found to have a piece of metal of unknown origin wedged between two phases. The safety evaluation allowed for continued operation until such time the metal could be removed.

It was concluded from the safety evaluation that the design safety function was not affected by the lodged metal fragment in the feeder cable and would not create or increase the probability of occurrence of a design basis accident.

2.6.4.25
PER 293-1203

This PER documented a situation where a steam leak was identified at the pipe-to-valve weld on the downstream side of Valve RCIC-V-26. The safety evaluation allowed for the on-line repair of the leak by means of Furmanite until permanent repairs could be made.

It was concluded from the safety evaluation that the proposed repair would not cause or contribute to any accidents, transients or special events as described in the LBDs.

2.6.4.26
PER 293-1217

This PER was written to address permanent removal of the top and bottom covers of the microprocessor assembly and relay tray in the control panels for CMS-CP-1301 and CMS-CP-1401 (H2/O2 Analyzer Control Units).

It was concluded from the safety evaluation that removal of the assemblies and relay trays would not result in a physical configuration which could affect other equipment. Furthermore, this activity neither alters nor increases the probability of equipment malfunction.

2.6.4.27
PER 293-1273

This PER was written to address continued operation with power cable trays covered by Thermo-Lag or a metal tray cover without the supporting cable ampacity calculations having been completed.

It was concluded from the safety evaluation that safety-related and safe-shutdown cables routed in raceways with tray covers installed are reasonably assured to perform their intended safety function even though formal ampacity calculations have not been completed. Furthermore, fire tours were in place.

2.6.4.28
PER 293-1308

This PER was written to address the removal of up to 1/2 inch of the stem on the Magnetrol float switches for RCIC-LS-15A and RCIC-LS-15B. This would allow the switches that shift RCIC suction from the Condensate Storage Tanks (CSTs) to the Suppression Pool to reset upon increasing level.

It was concluded from the safety evaluation that, since the trip setpoint of the switches would remain unaffected and the switches would remain fully functional, there would be no increase in the consequences of an accident previously evaluated.

2.6.4.29
PER 293-1319

This PER was written to address changing the stroke time of Relay SW-RLY-V/2B4 to compensate for a misadjustment of the closed limit switch.

It was concluded from the safety evaluation that the Service Water System would remain operable and capable of providing its safety function. Furthermore, no mechanism exists whereby delay of SW-V-2B opening time could increase the probability of an accident evaluated in the LBDs.

2.6.4.30
PER 293-1366

This PER documented a situation where, during the performance of a Technical Specification surveillance, Containment Atmosphere Control (CAC) System Valve CAC-V-4 failed to electrically stroke open. The safety evaluation allowed for modification of the manual declutch lever for CAC-V-4.

It was concluded from the safety evaluation that the re-orientation of the manual declutch lever met design and construction standards applicable to the component. Furthermore, the consequences of any accident evaluated previously would not be increased because the modification would not affect the design safety function of the valve.

2.6.4.31
PER 293-1378

This PER documented a situation where, during the performance of a surveillance test, Valve RCIC-V-63 did not meet its closing time limit of 10.0 seconds as specified by the FSAR. The safety evaluation allowed for extending the stroke time to 16 seconds.

It was concluded from the safety evaluation that even with the extended stroke time, all equipment remained qualified to perform the required safety function as defined in the LBDs. Therefore, the consequences of an accident previously evaluated would not change.

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 1993.

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 1993.

2.6.6.1 Procedure Revision Forms for PPMS 10.27.58A-D/10.27.61A-D

Several procedures were developed to instruct I&C Technicians in the backfilling of RPV level and pressure instrument sensing lines to remove entrapped air. Backfilling is a maintenance activity that is performed to ensure that the instrument line is filled with water and that any air or noncondensible gases are removed from the line and replaced with demineralized water.

It was concluded from the safety evaluation that these activities do not increase the probability of occurrence of an accident or increase the consequences of an accident previously evaluated in the LBDs. An analysis was performed to ensure that, despite a failure of the instrument line, the FSAR (Chapter 15) analyses were bounding and no consequences would result that were beyond the capability of operators or safety systems.

2.6.6.2 Procedure Deviation Form 93-277 for PPM 8.3.4

The procedure governing the Nondestructive Testing and Examination Program was changed to reflect results the annual review of existing NDE&I instructions.

It was concluded from the safety evaluation that the proposed activity would not increase the consequences of an accident previously evaluated in the LBDs. The procedural changes improve current NDE&I examination techniques.

2.6.6.3 Procedure Revision Form for PPM 6.5.16

This procedure was developed to provide instructions for Reactor Recirculation (RRC) System Jet Pump removal and installation.

It was concluded from the safety evaluation that the proposed activity of removal of a jet pump during shutdown (including dropping during removal) conditions is not an initiating event for any design basis accident. Therefore, the activity would not increase the probability of occurrence of an accident evaluated previously in the LBDs.

2.6.6.4
Procedure Deviation Form 93-582 for PPM 2.4.2

The procedure for operation of the Residual Heat Removal (RHR) System was modified to include a section to allow for the return of water to the Reactor Pressure Vessel through RHR-V-53B, rather than the Fuel Pool Cooling return line.

It was concluded from the safety evaluation that the change in the return path would not increase the consequences of either a fuel handling accident or a moderate-energy line break in the shutdown cooling system.

2.6.6.5
Procedure Revision Form for PPM 8.3.286

This procedure was developed for the performance of the preoperational test of the Reactor Building Stack Effluent Radiation Monitoring System (PRM-RE-1A, PRM-RE-1B, PRM-RE-1C and associated equipment).

It was concluded from the safety evaluation that there were no systems, structures or components important to safety that would be affected by performance of the test. Accordingly, the proposed activity would not increase the probability of occurrence of malfunction of equipment previously evaluated in the LBDs.

2.6.6.6
Procedure Revision Form for PPM 8.3.287

This procedure was developed to verify the design configuration of the continuous backfill from the Control Rod Drive (CRD) System. Performance of the procedure will provide test data to determine hydraulic transient characteristics.

It was concluded from the safety evaluation that all plant systems and components required to mitigate the consequences of accidents previously evaluated would be unaffected by performance of the procedure.

2.6.6.7
<u>Procedure Deviation Form 93-929 for PPM 7.4.0.5.16</u>

This procedure was changed to revise the normal lineup for Pump RHR-P-2A and Valve RHR-V-178A.

It was concluded from the safety evaluation that isolation of the keep-full subsystem would not impact the ability of the Standby Service Water (SSW) System to perform an event mitigation function. Accordingly, the consequences of an accident previously evaluated in the LBDs would not be increased.

2.6.6.8
<u>Procedure Deviation Form 93-930 for PPM 7.4.0.5.17</u>

This procedure was changed to revise the normal lineup for Pump RHR-P-2B and Valve RHR-V-178B.

It was concluded from the safety evaluation that isolation of the keep-full subsystem would not impact the ability of the SSW System to perform an event mitigation function. Accordingly, the consequences of an accident previously evaluated in the LBDs would not be increased.

2.6.6.9
<u>Procedure Deviation Form 93-931 for PPM 2.7.1B</u>

This procedure was modified to change the breaker lineup for SSW System Pumps SW-P-2A and SW-P-2B from "on" to "off".

It was concluded from the safety evaluation that isolation of the keep-full subsystem would not impact the ability of the SSW System to perform an event mitigation function. Accordingly, the consequences of an accident previously evaluated in the LBDs would not be increased.

2.6.6.10
<u>Procedure Deviation Form 93-1165 for PPM 2.3.5</u>

This procedure was changed to include a statement in the standby lineup section indicating that Standby Gas Treatment (SGT) System Valves SGT-V-2A and SGT-V-2B need to be failed open if the opposite train is inoperable.

It was concluded from the safety evaluation that the possibility of a system failure resulting from these valves failing to open upon demand is eliminated by the valves being placed in the normally-open position.

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 one valid failure in 1993 for the three emergency diesel generators.

- Identity of Diesel Generator and date of failure:

  Division One Emergency Diesel Generator (DG-1); June 6, 1993 (0400 Hours).

- Number designation of failure in last 100 valid tests:

  This was the first failure of the last 100 tests. The test was determined to be a "Valid" failure.

- Cause of failure:

  During the performance of a Technical Specification surveillance test, the Division One Emergency Diesel Generator failed to start upon a simulated LOCA actuation. During troubleshooting efforts, it was determined that the cause was due to a control relay (K16) that failed to change state.

- Corrective measures taken:

  The failed component was replaced and no further problems were identified.

- Length of time the Diesel Generator unit was unavailable:

  The Diesel Generator was out of service for approximately 13 hours and was returned to service at 1650 hours on June 6, 1993.

- Current surveillance test interval:

  Thirty-one days.

2.8 SEALED SOURCE CONTAMINATION

This section of the report contains information pertaining to sealed source contamination in accordance with WNP-2 Technical Specification 4.7.5.3. This specification requires a report to be submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

On October 27, 1993 at 1515 hours, a 1.4 millicurie source of Sr-90 (Serial Number WNP-2-79-042) was found to have 0.045 microcuries of removable contamination, exceeding the Technical Specification limit of 0.005 microcuries. The contamination was discovered during the performance of the semi-annual leak test of sealed radioactive sources in accordance with Plant Procedure (PPM) 11.2.14.7, "Leak Testing of Radioactive Sources."

Immediate corrective action consisted of removing the source from service and performing an isotopic evaluation of the smear. In addition, Problem Evaluation Request (PER) 293-1272 was written to document the identification of the source contamination. The probable cause for the contamination was abrasion of the source surface due to contact between the source and one or more of the shields contained in the source holder.

Further corrective action consisted of disposal of the source and holder as radioactive waste. An inspection was also performed on all other source holders of a similar design and no further deficiencies were identified.