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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352-0968 • (509) 372-5000

March 1, 1993
G02-93-048

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
ANNUAL OPERATING REPORT 1992

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 1992. Should you have any questions or comments, please contact Mr. A. G. Hosler, Manager, WNP-2 Licensing.

Sincerely,

JW Baker

J. W. Baker
WNP-2 Plant Manager (Mail Drop 927M)

Enclosure

cc: Mr. J. B. Martin, NRC - Region V
Mr. D. L. Williams, BPA (MD 399)
Mr. R. F. Mazurkiewicz, BPA (MD 399)
NRC Resident Inspector (MD 901A)

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WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1992

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

TABLE OF CONTENTS

1.0 INTRODUCTION

The 1992 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.

During January 1992 new monthly records were set for electricity generation and plant capacity factor when the plant ran, on average, at 100 percent power. On February 25, 1992 the plant was shutdown due to problems identified in the Containment Atmosphere Control (CAC) System. From an engineering analysis, it was determined that drain lines associated with the system were required to be modified to ensure the lines would remove water formed in the hydrogen recombiners. Following modification efforts and a final series of tests, the plant resumed full power operation on March 19, 1992.

On April 18, 1992 the plant was shutdown for the annual maintenance and refueling outage. In the remaining months of the year following the outage, the plant experienced three forced shutdowns due to 1) the loss of the "B" phase signal from a 500KV transformer potential device, 2) high drywell leakage, 3) core power oscillations of 20 percent power during preparation to change the Reactor Recirculation (RRC) System pumps to high-speed (60Hz) operation.

During 1992 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 seventh refueling outage was successfully completed and significant activities included:

- Replacement of the low-pressure Main Turbine Rotors (this was the culmination of a three-year, $30 million project that will reduce turbine maintenance and increase electrical output by an additional 15-to-20 megawatts).

- Off-loading of all 764 fuel assemblies and draining of the vessel to allow for inspection of reactor components and chemical decontamination.

- Chemical decontamination of the Reactor Recirculation (RRC) System.

- Rebuilding 30 of the Control Rod Drive Mechanisms.

- Nondestructive examination of Main Condenser tubes.

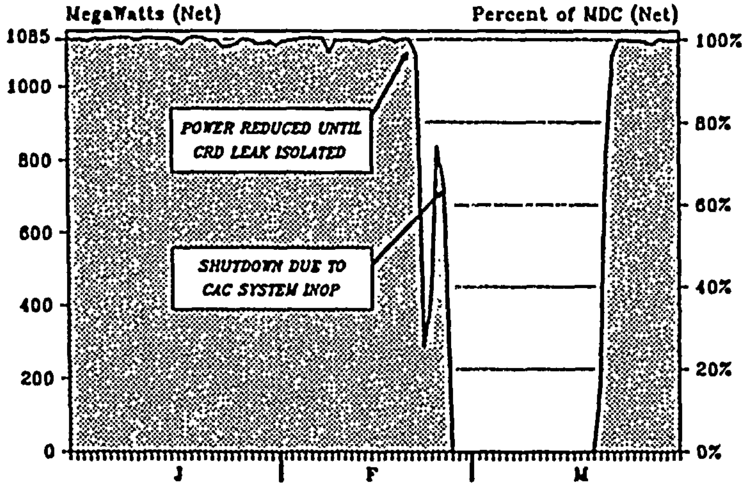
The 1992 capacity factors, based on net electrical energy output are listed below.

Month	Capacity Factor
January	100.18
February	78.84
March	37.34
April*	55.16
May	0
June**	0
July	25.68
August	37.01
September	95.78
October	99.82
November	86.91
December	102.95
Overall	59.73

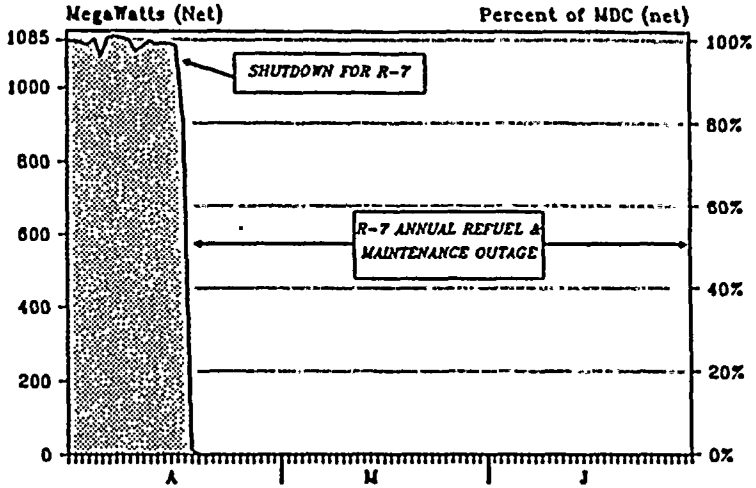
* Started Maintenance and Refueling Outage
** Ended Maintenance and Refueling Outage

NOTE: Capacity factors for 1992 were based on a Maximum Dependable Capacity (MDC) of 1085 MWe. (The MDC was revised to 1112 MWe in January 1993.)

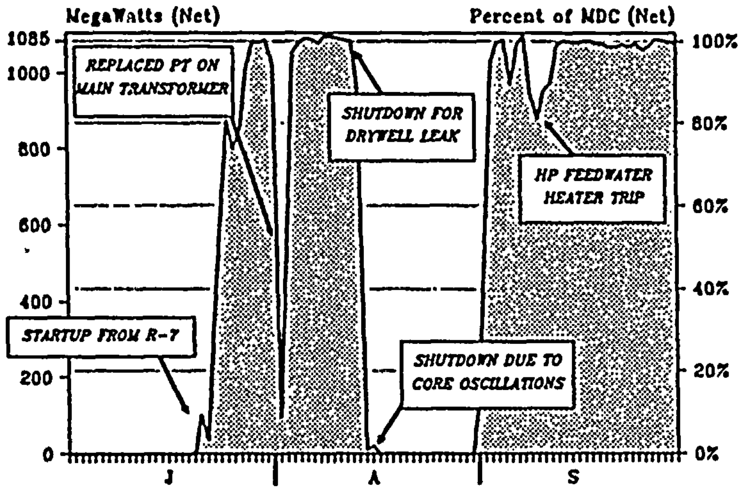
WNP-2 LOAD PROFILE - CALENDAR YEAR 1992



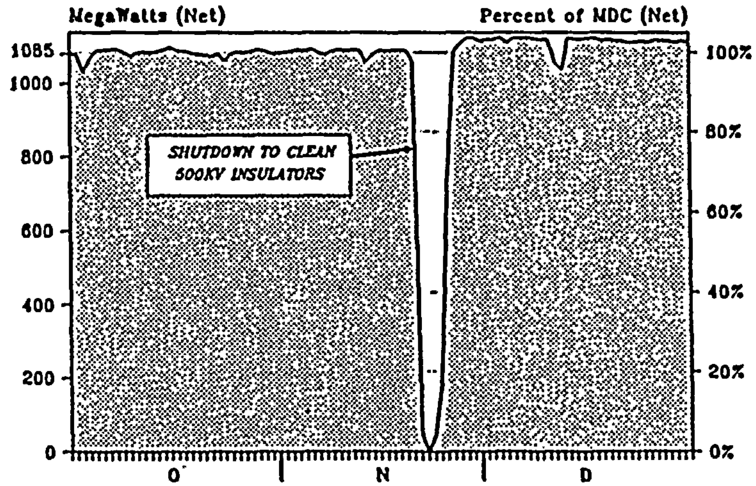
1st QUARTER 1992



2nd QUARTER 1992



3rd QUARTER 1992

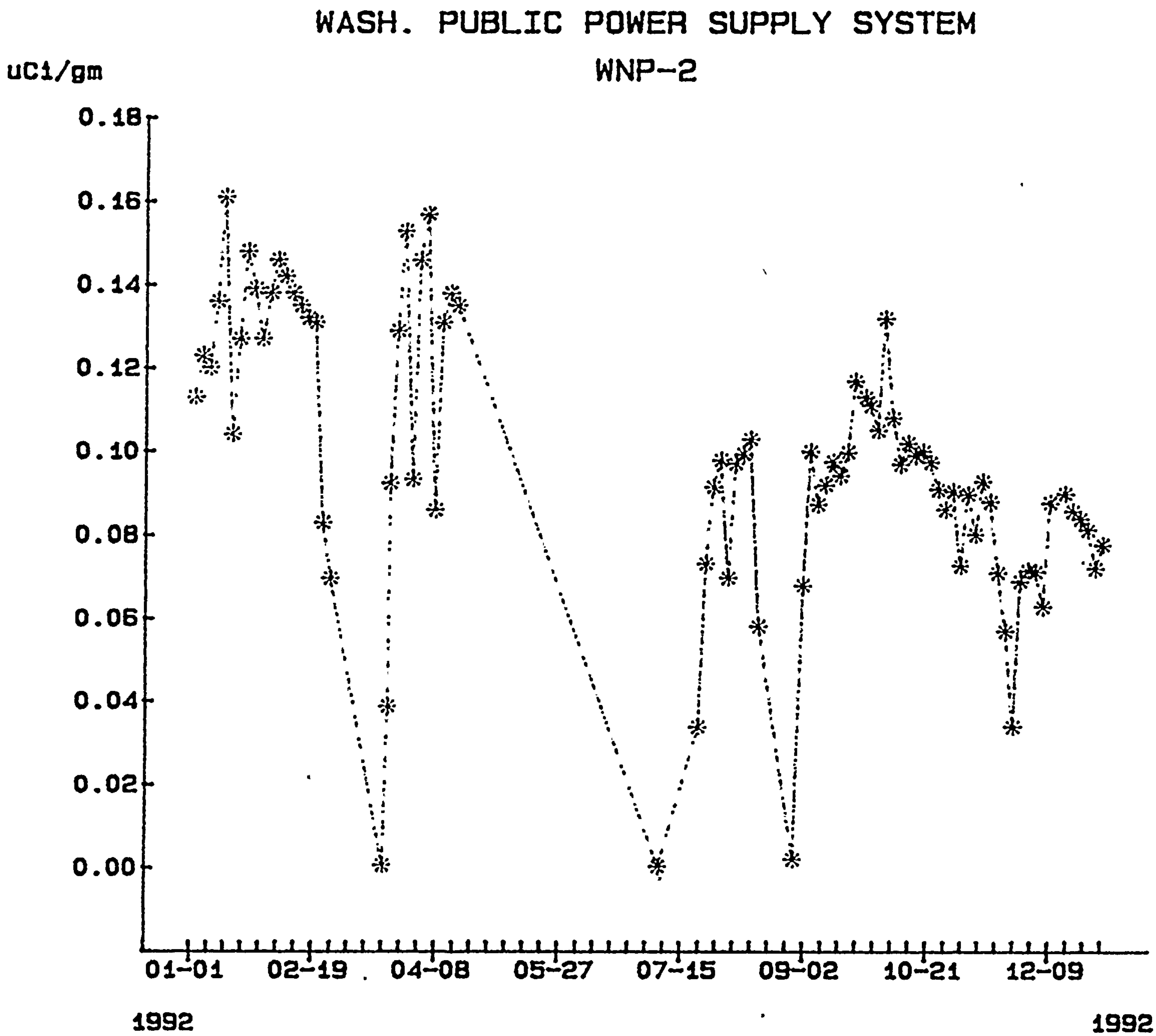


4th QUARTER 1992

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.

RER-020 WASHINGTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT / 1.16 APPENDIX A 01/26/93 10:51

NUCLEAR PLANT NO. 2		NUMBER OF PERSONS RECEIVING OVER 100 MREM			REPORT FOR CALENDAR YEAR 1992 TOTAL MAN-REM		
		STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS
OPERATIONS & SURVEILLANCE	MAINTENANCE PERSONNEL	2.999	0.000	1.630	3.310	0.000	1.185
	OPERATING PERSONNEL	1.629	0.000	0.000	1.556	0.000	0.000
	HEALTH PHYSICS PERSONNEL	2.904	0.000	0.255	2.562	0.000	0.306
	SUPERVISORY PERSONNEL	1.419	0.000	0.500	0.600	0.000	0.100
	ENGINEERING PERSONNEL	2.079	3.872	2.327	0.572	1.400	0.645
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	212.738	4.844	374.653	165.769	1.947	230.415
	OPERATING PERSONNEL	40.675	4.049	0.000	40.635	0.735	0.000
	HEALTH PHYSICS PERSONNEL	43.079	0.000	66.501	31.459	0.000	46.464
	SUPERVISORY PERSONNEL	21.337	1.000	2.631	7.016	0.503	0.905
	ENGINEERING PERSONNEL	28.515	34.044	62.759	11.092	11.067	38.143
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	0.361	0.120	2.322	0.395	0.095	1.227
	OPERATING PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.034	0.000	0.351	0.025	0.000	0.185
	SUPERVISORY PERSONNEL	0.245	0.000	0.000	0.070	0.000	0.000
	ENGINEERING PERSONNEL	0.203	0.577	0.176	0.185	0.191	0.110
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	8.705	0.529	12.056	5.990	0.420	4.758
	OPERATING PERSONNEL	0.444	0.000	0.000	0.462	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.708	0.000	0.399	0.929	0.000	0.355
	SUPERVISORY PERSONNEL	0.991	0.000	0.000	0.322	0.000	0.000
	ENGINEERING PERSONNEL	1.451	1.343	0.804	0.680	0.587	0.462
WASTE PROCESSING	MAINTENANCE PERSONNEL	2.058	0.038	2.651	2.470	0.030	0.937
	OPERATING PERSONNEL	0.020	0.000	0.000	0.020	0.000	0.000
	HEALTH PHYSICS PERSONNEL	3.538	0.000	2.590	3.513	0.000	2.434
	SUPERVISORY PERSONNEL	0.026	0.000	0.335	0.020	0.000	0.260
	ENGINEERING PERSONNEL	0.128	0.529	0.866	0.035	0.140	0.195
REFUELING	MAINTENANCE PERSONNEL	18.345	0.607	0.355	24.958	0.616	0.110
	OPERATING PERSONNEL	2.289	0.951	0.000	3.416	0.155	0.000
	HEALTH PHYSICS PERSONNEL	0.738	0.000	6.701	1.145	0.000	2.741
	SUPERVISORY PERSONNEL	1.815	0.000	0.025	0.660	0.000	0.005
	ENGINEERING PERSONNEL	1.813	1.797	0.521	0.935	1.260	0.310
TOTAL	MAINTENANCE PERSONNEL	245.206	6.138	393.667	202.892	3.108	238.632
	OPERATING PERSONNEL	45.057	5.000	0.000	46.089	0.890	0.000
	HEALTH PHYSICS PERSONNEL	51.001	0.000	76.797	39.633	0.000	52.505
	SUPERVISORY PERSONNEL	25.833	1.000	3.491	8.688	0.503	1.270
	ENGINEERING PERSONNEL	34.189	42.162	67.453	13.499	14.645	39.865
GRAND TOTAL		401.286	54.300	541.408	310.801	19.146	332.272

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 1992 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	63790-00-0124	63790-00-0061			
Component ID (Location)	MS-RV-2D	MS-RV-5B			
Date of Actuation (MO/DA/YR)	03/19/92	03/19/92			
Time of Day (24 Hour Clock)	0228	0217			
Type of Actuation (Code)	B	B			
Cause/Reason for Actuation (Code)	C	C			
Rx Operating Condition Prior to Lift (Code)	C	C			
Rx Power Level Prior to Lift (% Rated Thermal)	15%	15%			
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A			
Other Instrumentation Type (Code)	PROCESS COMPUTER	PROCESS COMPUTER			
Other Instrumentation Number Reading and Units	Open	Open			
Rx Pressure Prior to Actuation (PSIG)	943	943			
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A			
Duration of This Actuation (Minutes, Seconds)	30 sec	1 min, 25 sec			
Failures, Reports (Code)	C	C			
LER Number (5 Digit Number)	None	None			
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes			

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

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0122	63790-00-0126	63790-00-0047	63790-00-0058	
Component ID (Location)	MS-RV-1D	MS-RV-3D	MS-RV-2C	MS-RV-4C	
Date of Actuation (MO/DA/YR)	04/18/92	04/18/92	04/18/92	04/18/92	
Time of Day (24 Hour Clock)	1308	1330	1409	1425	
Type of Actuation (Code)	C	C	C	C	
Cause/Reason for Actuation (Code)	C	C	C	C	
Rx Operating Condition Prior to Lift (Code)	D	D	D	D	
Rx Power Level Prior to Lift (% Rated Thermal)	0%	0%	0%	0%	
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	
Other Instrumentation Type (Code)	A	A	A	A	
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	
Rx Pressure Prior to Actuation (PSIG)	914	936	935	936	
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	
Duration of This Actuation (Minutes, Seconds)	6 sec.	6 sec.	6 sec.	6 sec.	
Failures, Reports (Code)	C	C	C	C	
LER Number (5 Digit Number)	None	None	None	None	
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	

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

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0048	63790-00-0054	63790-00-0055	63790-00-0059	63790-00-0045
Component ID (Location)	MS-RV-1A	MS-RV-2A	MS-RV-3A	MS-RV-4A	MS-RV-1B
Date of Actuation (MO/DA/YR)	07/06/92	07/06/92	07/06/92	07/06/92	07/06/92
Time of Day (24 Hour Clock)	0338	0433	0404	0328	0423
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	28 sec	16 sec	48 sec	15 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)

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0049	63790-00-0052	63790-00-0056	63790-00-0061	63790-00-0046
Component ID (Location)	MS-RV-2B	MS-RV-3B	MS-RV-4B	MS-RV-5B	MS-RV-1C
Date of Actuation (MO/DA/YR)	07/06/92	07/06/92	07/06/92	07/06/92	07/06/92
Time of Day (24 Hour Clock)	0348	0251	0359	0415	0343
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)	19 sec	None	50 sec	28 sec	29 sec
Failures, Reports (Code)	C	B,D	C	C	C
LER Number (5 Digit Number)	None	92-033	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)

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0047	63790-00-0051	63790-00-0058	63790-00-0062	63790-00-0050
Component ID (Location)	MS-RV-2C	MS-RV-3C	MS-RV-4C	MS-RV-5C	MS-RV-1D
Date of Actuation (MO/DA/YR)	07/06/92	07/06/92	07/06/92	07/06/92	07/06/92
Time of Day (24 Hour Clock)	0430	0334	0408	0320	0412
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)	24 sec	32 sec	33 sec	45 sec	15 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)

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-0124	63790-00-0126	63790-00-0060	63790-00-0052	63790-00-0052
Component ID (Location)	MS-RV-2D	MS-RV-3D	MS-RV-4D	MS-RV-3B	MS-RV-3B
Date of Actuation (MO/DA/YR)	07/06/92	07/06/92	07/06/92	07/06/92	07/11/92
Time of Day (24 Hour Clock)	0419	0351	0312	0455	1335
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	940	888
Duration of This Actuation (Minutes, Seconds)	19 sec	1 min, 21 sec	1 min, 45 sec	2 min, 29 sec	2 min, 58 sec
Failures, Reports (Code)	C	C	C	B,D	B,D
LER Number (5 Digit Number)	None	None	None	92-033	92-033
Comments Regarding This Actuation Attached? (Yes or No)	Yes	Yes	Yes	Yes	Yes

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

NOTE: Includes all In Situ Tests

For Each Actuation or Failure to Actuate:	1.	2.	3.	4.	5.
S/R Valve Serial Number	63790-00-053	63790-00-0060	63790-00-0051	63790-00-0049	63790-00-0045
Component ID (Location)	MS-RV-3B	MS-RV-4D	MS-RV-3C	MS-RV-2B	MS-RV-1B
Date of Actuation (MO/DA/YR)	07/19/92	07/19/92	07/19/92	07/19/92	07/19/92
Time of Day (24 Hour Clock)	2007	2018	2025	2030	2034
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)	948	948	948	948	948
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)	45 sec	1 min	11 sec	10 sec	16 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)

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

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

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: **FEBRUARY 1992**

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
92-01	2/22/92	F	14.2	A	1		RB	CRDRVE	The plant was downpowered and generator was removed from grid to permit drywell entry for verification of source of FDR leakage. It was found to be coming from flange of CRD 42-59. The CRD was isolated and the plant returned to power operation.
92-02	2/25/92	F	101.3	H	1		SE	RECOMB	The plant was shutdown after an engineeringing evaluation determined that drain piping from both Containment Atmiohsphere Control (CAC) units was improperly designed. Modification of drain piping is in progress.

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)

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: MARCH 1992

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
92-02	2/25/92	F	436.4	H	4	92-007	SE	RECOMB	Concluded outage for modificaiton of Containment Atmospheric Control (CAC) drain piping.

SUMMARY: WNP-2 returned to service on March 19 after completion of plant modifications.

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)

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: APRIL 1992

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
92-03	4/18/92	S	308.4	C	1	--	RC	FUEL XX	Plant was shutdown as scheduled for refueling outage R-7.

SUMMARY: WNP-2 operated routinely until April 18, 1992 when it was shutdown for refueling outage R-7.

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)

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: JULY 1992

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
92-03	4/18/92	S	460.4	C	4	--	RC	FUEL XX	Concluded refueling outage R-7.
92-04	7/20/92	S	3.7	B	1	--	HA	MECFUN	Generator was removed from service for overspeed testing and turbine bypass valve testing. It was resynchronized to grid after successful completion of testing.
92-05	7/21/92	S	16.4	B	1	--	HA	TURBIN	Generator was removed from grid for torsional testing of turbine rotors. After satisfactory completion of testing, it was returned to service.

SUMMARY: WNP-2 returned to service from refueling. Subsequently, two scheduled outages for testing were completed.

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 &		5-REDUCED LOAD	LICENSEE EVENT REPORT
	LICENSE EXAM		9-OTHER	(LER) FILE (NUREG-0161)

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: AUGUST 1992

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURRENCE
92-06	8/1/92	F	12.95	A	1	--	EB	TRANSF	Generator was removed from BPA grid due to loss of signal from B Phase 500KV potential device. The bushing was replaced and generator resynchronized to grid.
92-07	8/14/92	F	19.0	A	1	--	CG	VALVEX	The reactor was downpowered and generator removed from grid to permit drywell entry for identification and repair of drywell leakage. The leakage was identified as a packing leak on RWCU-V-103. The valve was manually backseated to stop leak and the plant returned to power operation.
92-08	8/15/92	F	385.56	A	2	92-037	RC	FUELXX	The reactor was manually scrammed from 35% power due to core instability. Core power oscillations of 20% power occurred during preparation to change RRC pumps to high speed (60 Hz) operation. After investigation and evaluation of the event by an NRC Augmented Team, a number of recommendations and corrective actions were implemented and the plant returned to power operation on August 31, 1992 (see LER 92-037).

SUMMARY: WNP-2 incurred three forced outages in August 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)

UNIT SHUTDOWNS / REDUCTIONS

REPORT PERIOD: NOVEMBER 1992

NO.	DATE	TYPE	HOURS	REASON	METHOD	LER NO	SYSTEM	COMPONENT	CAUSE & CORRECTIVE ACTION TO PREVENT RECURREN
92-09	11/21/92	S	53.8	B	1		EB	ELECON	Generator was removed from service for cleaning of high voltage insu due to buildup of chemical deposits from cooling tower drift. Als repaired source of Drywell FDR leakage (RFW-V-10B). The plan then returned to service.

SUMMARY:

WNP-2 incurred one scheduled outage in November 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 (CONTINUED)

GENERATOR RUN TIME - 1992

DATE	GENERATOR ON LIN	GENERATOR OFF LIN	NO. S.D.	COMMENTS	HOURS RUN TIME	OUTAGE
2/22/92		1312 (F)	1 110	PLANT WAS DOWNPOWERED AND GENERATOR REMOVED FROM GRID AT 1312 HOURS ON 2/22 TO PERMIT DRYWELL ENTRY IN SEARCH OF LEAKAGE. ALSO CRD 42-59 STARTED DRIFTING IN AND CONTAINMENT PARTICULATE MONITORS CMS-RIS-12/1A & B INCREASED FROM 600 TO 1800 CPM IN 30 MINUTES. FDR DRYWELL LEAKAGE WAS 2.22 GPM BY ACTUAL MEASUREMENT. UPON ENTRANCE OF DRYWELL, THE SOURCE OF LEAKAGE WAS FOUND TO BE FROM CRD 42-59 FLANGE. THE CRD WAS ISOLATED TO STOP LEAK.	(CY 92) 1261.20 (TOT RUN) 1404.12	
2/23/92	0322			GENERATOR SYNCHRONIZED TO BPA GRID.		14.17
2/25/92		1838 (F)	2 111	AN UNUSUAL EVENT WAS DECLARED ON 2/25 AT 1315 HOURS AS A RESULT OF A DISPOSITION BY POC ON PER 292-0150 THAT BOTH CAC SYSTEMS ARE NOT DESIGNED PROPERLY AND ARE THEREFORE INOPERABLE. PLANT SHUTDOWN WAS COMMENCED AND THE GENERATOR WAS TAKEN OFF LINE AT 1838 HOURS.	63.26	
3/19/92	0425			GENERATOR SYNCHRONIZED TO BPA GRID.		537.79
4/18/92		0335	3 112	GENERATOR WAS REMOVED FROM GRID AT 0335 HOURS ON 4/18 FOR START OF R-7 REFUELING OUTAGE.	718.16	
7/20/92	0423			R-7 OUTAGE OFFICIALLY ENDED WITH FIRST CLOSING OF GENERATOR BREAKER AT 0423 ON 7/20.		2,232.80
7/20/92		1955	4 113	GENERATOR WAS REMOVED FROM GRID AT 1955 HOURS ON 7/20 FOR OVERSPEED TESTING.	15.53	
7/20/92	2335			GENERATOR SYNCHRONIZED TO BPA GRID.		3.67
7/21/92		0424	5 114	GENERATOR WAS REMOVED FROM GRID AT 0424 HOURS ON 7/21 FOR TURBINE TORSIONAL TESTING.	4.82	
7/21/92	2050			GENERATOR SYNCHRONIZED TO BPA GRID.		16.43
8/01/92		0505 (F)	6 115	GENERATOR WAS REMOVED FROM GRID DUE TO LOSS OF "B" PHASE SIGNAL FROM TR-M4 500kv BUSHING POTENTIAL DECVICE. BPA REPLACED IT WITH ONE FROM TR-M2 SPARE TRANSFORMER.	248.25	
8/01/92	1802			GENERATOR SYNCHRONIZED TO BPA GRID.		12.95
8/14/92		0209 (F)	7 116	GENERATOR WAS REMOVED FROM GRID DUE TO DRYWELL LEAKAGE EXCEEDING 3 GPM. SOURCE OF LEAK WAS FROM RWCU-V-103 PACKING.	296.12	
8/14/92	2109			GENERATOR SYNCHRONIZED TO BPA GRID.		19.00
8/15/92		0304 (F)	8 117	REACTOR WAS MANUALLY SCRAMMED AT 35% POWER DUE TO CORE POWER OSCILLATIONS OF 20% POWER. OSCILLATIONS WERE BETWEEN 25% & 45% ON APRM RECORDERS. ALSO NUMEROUS LPRM DOWNSCALES.	5.92	
8/31/92	0438			GENERATOR SYNCHRONIZED TO BPA GRID.		385.56
11/21/92		0529	9 118	GENERATOR WAS REMOVED FROM GRID FOR CLEANING OF SY INSULATORS & TO REPAIR LEAK IN DRYWELL. LEAK WAS FROM RFW-V-10B.	1969.85	
11/23/92	1115					53.77
				THRU 12/31/92	924.75	
				CYTD	5507.86	3,276.14

(F) - FORCED OUTAGE

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

This section of the report normally contains the information required in accordance with the Regulatory Guide 1.16, Revision 4, Section C.1.b(2)(e).

However, for this reporting period (1992), information on significant corrective maintenance performed on safety-related equipment will be submitted separately from this annual operating report.

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 7 was performed in the month of October 1992. The purpose of the inspection was to verify assembly and fuel rod integrity. The inspection was also used to fulfill a commitment to inspect Siemens Power Corporation (SPC) high-burn-up 8X8 assemblies and note any unusual fuel rod growth. At the same time, a visual inspection of two discharged fuel channels was also performed.

A total of four fuel assemblies and two channels discharged at the end of Cycle 7 were inspected. No evidence of rod bow, abnormal fuel rod growth or mechanical damage were noted during the inspection of the assemblies. Furthermore, little or no nodular corrosion was observed on the clad surface.

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 1992 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) or Basic Design Change (BDC). The following PMRs/BDCs implemented in 1992 required a Safety Evaluation in accordance with 10CFR50.59. Each permanent change was evaluated and determined neither to represent an Unreviewed Safety Question nor require a change to the WNP-2 Technical Specifications.

2.6.1.1
BDC 55-1999-OA

This BDC provided for the correction of plant drawings to reflect the as-built condition of Control Air System (CAS) valve CAS-V-100/93. During construction the valve was added to the plant, however, the top-tier drawing was not updated to reflect the existing configuration.

It was concluded from the safety evaluation that this activity was limited to as-built and documentation changes only to non-safety related components in a non-safety related system. Accordingly, this activity would not alter assumptions made previously in the Licensing Basis Documents (LBD) or impair any operator action.

2.6.1.2
BDC 91-0309-OA

This BDC provided for the replacement of spacers in the Standby Service Water (SSW) System at the inlet to Diesel Cooling Water (DCW) Heat Exchangers DCW-HX-1A2/1B2 with restricting orifices.

It was concluded from the safety evaluation that this activity would meet the design, material and construction standards applicable to the SSW System. Furthermore, the system would not be required to operate outside of its design limits and no changes to system interfaces were created by this activity.

2.6.1.3
BDC 91-0071-Z

This BDC provided for the modification of several test/vent/drain line connections. The scope of the BDC was to modify the connection configuration from socket welds to butt welds on several lines.

It was concluded from the safety evaluation that the change to the test/drain/sample/vent/instrument line would provide stronger welds and a stronger section of pipe more resistant to shock and less likely to fatigue and break than the original configuration. The proposed activity increased the endurance of the weld in the test/drain/sample/vent/instrument line.

2.6.1.4
BDC 55-2256-OA

This BDC provided for the clarification of Drawing M524 to allow the flexibility of using Standby Service Water (SSW) System valves SSW-V-70A/B during plant operations other than shutdown.

It was concluded from the safety evaluation that a malfunction of equipment which would require the use of the service water system and/or spray pond has been addressed in the LBD discussion of mitigating accidents. Furthermore, the use of these valves would not reduce the availability of the service water system or spray pond.

2.6.1.5
PMR 90-0268-OA

This PMR provided for the modification of the position indication mechanism for Reactor Core Isolation Cooling (RCIC) System testable check valve RCIC-V-66 on the head spray line by addition of a second return spring, a second set screw securing the movement arm cam to the indicator shaft, and removal of the unused switch trip pins on the movement arm cam.

It was concluded from the safety evaluation that this activity affected only the valve position indication mechanism, which has no safety function. In addition, the modification would improve the ability of the disk position switch trip arm to return to a position enabling correct valve position indication.

2.6.1.6
<u>PMR 86-0627-OA</u>

This PMR provided for installation of a cross-tie between the discharge of Fuel Pool Cooling (FPC) System pump FPC-P-3 and the Equipment Drains Radioactive (EDR) System. Installation of the cross-tie would allow the water volume in the suppression pool to be lowered using the FPC System.

It was concluded from the safety evaluation that the new configuration would eliminate the need to start Residual Heat Removal (RHR) System pump RHR-P-2B by using FPC-P-3, a non-safety related piece of equipment. Furthermore, it was concluded that the installation of the cross-tie and operation of associated piping and valves would not increase the probability of the occurrence of any accident evaluated in the LBD.

2.6.1.7
<u>BDC 89-0299-23AP</u>

This BDC provided for replacement of three Reactor Core Isolation Cooling (RCIC) System Bailey pressure transmitters (RCIC-PT-4, RCIC-PT-5 and RCIC-PT-7) with Rosemount Series 1153 transmitters.

It was concluded from the safety evaluation that the form and function were identical for both types of transmitters. In addition, it was concluded that the loop function would remain as originally designed, there was no change in system function and the modification does not increase the probability of an occurrence of an accident previously evaluated in the LBD.

2.6.1.8
<u>BDC 88-0048-OA</u>

This BDC provided for the replacement of Diesel Oil (DO) System Magnetrol capacitance level sensors and transmitters (DO-LITS-10A, DO-LITS-10B and DO-LITS-15) with Magnetrol ultrasonic level sensors and transmitters. In addition, local level indicators DO-LI-10A, DO-LI-10B and DO-LI-15 were to be replaced with Dixon Bar Graphs.

It was concluded from the safety evaluation that this activity serves to improve the reliability of the diesel oil measurement in the diesel oil storage tanks. Furthermore, it was concluded that there was no relationship between the effects of this activity and the probability of occurrence of any event described in Chapter 15 of the FSAR.

2.6.1.9
<u>BDC 87-0244-OA</u>

This BDC provided for installation of the Reactor Recirculation (RRC) System Adjustable Speed Drives (ASDs) and auxiliary equipment.

It was concluded from the safety evaluation that the change involves only the location and installation of the ASDs and auxiliary equipment and there are no direct connections to existing equipment or systems important to safety. Since the change only involves the location and installation of the ASDs, the safety evaluation concluded that no new failure mode for safety-related equipment or systems would be introduced by this installation.

2.6.1.10
<u>BDC 90-0100-OA</u>

This BDC provided for installation of pipe connections and blind flanges in four locations in the Scram Discharge Volume (SDV) System for the purpose of fresh water or chemical cleaning.

It was concluded from the safety evaluation that the modification would not affect the safety function of the SDV System and there would be no increase in the probability of occurrence of malfunction of equipment important to safety. It was also concluded that the proposed activity met the original design specification.

2.6.2 TEMPORARY MODIFICATIONS AND INSTRUMENT SETPOINT CHANGES

The following are summaries of temporary modifications and instrument 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 instrument setpoint changes are made under the Instrument Setpoint Change Request (ISCR) process.

2.6.2.1
TMR 92-082

This TMR provided for the installation of a regulating transformer in series with the alternating source circuit for Inverter IN-3 to limit fault current and provide fuse coordination in the event of a downstream short circuit.

It was concluded from the safety evaluation that the new circuit configuration was functionally identical to that of the original design and that installation of the regulating transformer would not increase the probability of occurrence of an accident previously evaluated in the LBD.

2.6.2.2
ISCR 1111

This ISCR provided for changing the instrument setpoints for the vacuum breakers between the Reactor Building and the Suppression Pool. Based on device uncertainties and manufacturer's range of operation, it was concluded that the maximum setting should be 0.443 psid instead of 0.50 psid.

It was concluded from the safety evaluation that establishment of the lower setpoint would not increase the probability of occurrence of an accident and the consequences of vacuum breaker opening is not impacted by this change.

2.6.2.3
<u>ISCR 1112</u>

This ICSR provided for changing the instrument setpoints for several Main Steam Leakage Control (MSLC) System pressure switches to allow initiation of the inboard MSLC System at greater than 35 psig containment pressure.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident and the only change is to allow initiation of the inboard system at peak containment pressure.

2.6.2.4
<u>ISCR 1121</u>

This ISCR provided for changing the instrument setoints for several Main Steam Leakage Control (MSLC) System pressure indicating switches to allow initiation of the inboard MSLC System at peak containment pressure.

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 only change is to allow initiation of the inboard MSLC System at a pressure between the Main Steam Isolation Valves (MSIVs) that is greater than peak containment pressure of 34.7 psig.

2.6.2.5
<u>ISCR 1122</u>

This ISCR provided for changing the instrument setpoint for Main Steam Leakage Control (MSLC) System pressure indicating switch MSLC-PIS-20 to allow initiation of the outboard MSLC System at peak containment pressure.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident and the only change is to allow initiation of the outboard MSLC System at peak containment pressure of 34.7 psig.

2.6.2.6
<u>ISCR 1120</u>

This ISCR provided for changing the instrument setpoint for Main Steam Leakage Control (MSLC) System pressure indicating switch MSLC-PIS-24 to allow initiation of the outboard MSLC System at peak containment pressure.

It was concluded from the safety evaluation that changing of the setpoint would not increase the probability of an accident and the only change is to allow initiation of the outboard MSLC System at peak containment pressure of 34.7 psig.

2.6.2.7
<u>ISCR 1150</u>

This ISCR provided for changing the intrument setpoint for Reactor Feedwater (RFW) System flow switches RFW-FS-618A and RFW-FS-618B in response to core flow instability issues raised following a scram that occurred on August 15,1992. The setpoint changes the Reactor Recirculation (RRC) System low feedwater flow cavitiation interlock to a lower value.

It was concluded from the safety evaluation that changing of the setpoint would neither increase the probability of the occurrence of an accident nor increase the probability of malfunction of equipment important to safety. Changing of the setpoint involves cavitation protection and there is sufficient design margin.

2.6.2.8
<u>ISCR 1160</u>

This ISCR provided for changing the instrument setpoints for several Radwaste Building Mixed Air (WMA-HVAC) System temperature switches to meet Station Blackout requirements that battery room temperature should be maintained $\geq$ 74 degrees F.

It was concluded from the safety evaluation that changing of the setpoints does not affect system function and the new setpoints were within the limits of the engineering design calculation.

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 91-058

This SCN provided for revision of the Emergency Plan in that certain tables and associated notes describing Emergency Action Level (EAL) initiating conditions were deleted from Section 6 of Chapter 13.3, "Emergency Preparedness Plan (EPP)."

It was concluded from the safety evaluation that elimination of certain EAL examples from the EPP would not impact the consequences of an accident or reduce evaluation for the margin of safety.

2.6.3.2
SCN 92-002

This SCN provided for clarification of the Residual Heat Removal (RHR) System cross-connect to the Fuel Pool Cooling (FPC) System.

It was concluded from the safety evaluation that the consequences of a malfunction would not increase and operation Loop B of RHR in the FPC assist mode would not cause an accident different from those previously analyzed.

2.6.3.3
SCN 92-036

This SCN provided for the revision of the Emergency Plan by deleting reference to use of the Department of Energy (DOE) helicopter for providing protective action notification to the Columbia River transient population within the WNP-2 Emergency Planning Zone (EPZ).

It was concluded from the safety evaluation that this activity would not impact the consequences of an accident because use of the helicopter was one of three actions taken to provide notification to a particular segment of the public. However, to compensate for the loss of the helicopter, the Supply System installed additional sirens to provide notification to the transient population within the EPZ for the Columbia River.

2.6.3.4
SCN 92-041

This SCN provided for the revision of the discussion of the Reactor Protection System (RPS) Motor-Generator (MG) Set coastdown time to reflect plant data.

It was concluded from the safety evaluation that the probability of transients resulting from power disturbances would not be increased because the MG Set maintains the one-second licensing basis and there is sufficient margin.

2.6.3.5
SCN 92-048

This SCN provided for revision of the description of activities on or near the site including construction of the Plant Engineering Center (PEC), various Hanford Site activities, air traffic patterns, Yakima Firing Range activities and gas pipelines.

It was concluded from the safety evaluation that the proposed activities would not increase the probability of occurrence of an accident or create the possibility of an accident of a different type than previously evaluated in the LBD as those activities that did involve some hazard were too distant from the site.

2.6.3.6
SCN 92-055

This SCN provided for partial implementation of the WNP-2 Cycle 8 core thermal limits and other core-related changes.

It was concluded that the Cycle 8 reload design would not increase the probability of occurrence of previously evaluated accidents or introduce any new equipment malfunctions.

2.6.3.7
SCN 92-060

This SCN provided for the revision of the Emergency Plan by deleting reference to the Crisis Management Center (CMC) as an emergency facility and the Managing Director Representative position.

It was concluded that the proposed activity would have no impact on occurrence of an accident and deletion of the CMC and Managing Director Representative position would not increase the consequences of an accident.

2.6.3.8
SCN 92-072

This SCN provided for the revision of FSAR Sections 13.1.1 and 13.1.2 to reflect several organizational changes that were made.

It was concluded from the safety evaluation that the proposed activity would not change the intent or commitments described in the LBD.

2.6.3.9
SCN 92-078

This SCN provided for revision to Diesel Generator loading schedules and associated text in FSAR Section 8.3.

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 92-091

This SCN provided for revision to FSAR Section 9.2.5 to describe the results of a re-evaluation of the WNP-2 Ultimate Heat Sink Analysis.

It was concluded from the safety evaluation that the revised Ultimate Heat Sink (UHS) Analysis shows that the UHS would be capable of accomplishing its safety function following a LOCA, without the availability of offsite power, and that the cooling capability would be maintained for 30 days without outside makeup.

2.6.4 PROBLEM EVALUATIONS

The Plant Problems-Plant Problem Reports Procedure (PPM 1.3.15) provides 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 291-0874

This PER was written because directions provided to change the plant emergency DC battery charger current limit settings were not made in accordance with an approved process. Directions were provided to change the current limit settings from 125 percent to 110 percent. However, these chargers were tested during startup with current limiting settings at 125 percent. Following further review, it was determined that there was not a problem with revising the charger current limit from 125 percent to 110 percent of rated output. An evaluation was performed which determined that there was sufficient margin in the capacity of the chargers to restore the associated battery from minimum charge state to a fully charged state within 24 hours while supplying the continuous loads connected to the distribution bus.

It was concluded from the safety evaluation that the loss of charger function is not an accident initiator and would not increase the probability of an accident previously evaluated in the LBD. Furthermore, revising the charger current limit setting would not increase the consequences of an accident evaluated previously as the safety function to supply the connected load while recharging the associated station battery remains the same.

2.6.4.2
PER 292-0171

This PER documented a situation where it was discovered that surveillance procedures for the Standby Gas Treatment System (SBGT) had flow readings being taken in CFM without any compensation for density variations. Analysis had shown that flow under harsh environment conditions would be high enough to trip the overloads on system fans. Following further evaluation, the SBGT flow limiter setpoint was revised.

It was concluded from the safety evaluation that consequences of an accident were not impacted and the new flow limiter setpoint would ensure that the SBGT System would operate within its design envelopes.

2.6.4.3
PER 292-0193

This PER documented a situation where it was determined that Main Steam Leakage Control (MSLC) System 50-minute timer MSLC-RLY-TK/2 may not allow enough time for the outboard main steam lines to depressurize. Following further evaluation, procedures were revised to allow the timer to be reset which would allow adequate time for depressurization.

It was concluded from the safety evaluation that the procedural changes would not increase the probability of occurrence of an accident previously identified. Furthermore, even with the extended depressurization period, all equipment would be bounded by previous analysis and, therefore, there would be no increase in the consequences of an accident evaluated previously in the LBD.

2.6.4.4
PER 292-0229

This PER was written to document Containment Atmosphere Control (CAC) System operability concerns pertaining to 1) automatic versus manual recycle valve control, 2) controlling CAC recycle flow based on recombiner exit temperature, and 3) validity and accuracy of CAC flow meter readings. Following further evaluation, actions taken included 1) the change of recycle valve control to manual at a constant recycle ratio rather than manually varying the recycle valve to maintain catalyst bed temperatures below 1150 degrees F, 2) changing Service Water (SW) scrubber flow strategy, and 3) considering a change in recycle flow ratio during the long-term, post-accident time period.

It was concluded from the safety analysis that the actions taken would ensure equipment operability during accident conditions such that the consequences of an accident would be mitigated. Furthermore, analysis results demonstrated that the CAC System would function to maintain oxygen levels inside containment below the levels originally reported in the FSAR.

2.6.4.5
PER 292-0287

This PER documented a situation where it was determined that the Appendix R calculation required certain operator actions to be taken in the event of fires both within the main control room and outside of the main control room. However, these areas had not been provided with the required emergency lighting. Following further evaluation, the decision was made to use portable battery-powered lighting in place of permanently-installed, battery-powered lamps.

It was concluded from the safety analysis that the use of portable lights would not increase the consequences of an accident since they are mitigation equipment. Furthermore, the portable lighting was shown, by a plant walkdown of all actions required and areas involved, to provide adequate lighting to allow for the performance of necessary actions.

2.6.4.6
PER 292-0596

This PER documented a situation where, during the maintenance and refueling outage, an object believed to be a "Brag Rag" was observed to be floating around in the vessel about two-thirds of the way down from the pool surface to the core top. Efforts to retrieve the object were unsuccessful.

It was concluded from the safety evaluation that damage to fuel caused by fuel assembly flow blockage, interference with control rod operation and harmful chemical impacts from the "Brag Rag" were considered to be insignificant.

2.6.4.7
PER 292-0702

This PER documented a situation where it was discovered that the as-found settings for the Emergency Diesel Generator Overspeed Trip Time Delay in the electric overspeed trip circuit did not agree with the design settings. Following further evaluation, the decision was made to continue to operate with the time delay relay settings different than indicated on plant drawings until they could be recalibrated.

It was concluded from the safety evaluation that no increase in equipment malfunction would occur with the existing settings because all time delay relay settings in the field were conservative to the required safety functions. Furthermore, all the relays identified provide accident mitigation only (or none at all) and, therefore, changes in the settings would not increase the probability of occurrence of an accident.

2.6.4.8
PER 292-0749

This PER documented a situation where, during maintenance for Reactor Feedwater (RFW) System flow control valve RFW-FCV-10B, damage to the valve internals was noted. Following further review, the decision was made to continue operation until a new shaft and plug assembly could be obtained and installed.

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

degraded condition of the flow control valve. Furthermore, failures associated with this component remain bounded and would not exceed or change previously evaluated limiting conditions analyzed.

2.6.4.9
PER 292-0789

This PER documented a situation where switchgear breaker control circuits for Reactor Closed Cooling (RCC) Water System pumps, Reactor Building Outside Air (ROA-HVAC) System fans and Reactor Building Exhaust Air (REA-HVAC) System fans were not provided with proper electrical separation. Following further review, the affected areas were placed on an hourly fire tour until the design discrepancies could be solved.

It was concluded from the safety evaluation that, since compensatory conditions were in place, the resulting conditions were equivalent to proper application of electrical separation practices. The compensatory measures consisted of verification of the area fire detection system component operability and an hourly fire tour of the affected areas.

2.6.4.10
PER 292-0871

This PER documented a situation where it was determined that Radwaste Building Mixed Air (WMA-HVAC) fans WMA-FN-52A and WMA-FN-52B would trip on Loss of Offsite Power (LOOP) and not automatically restart. These fans provide cooling to the cable spreading room. Following further review, procedures were modified to require a manual restart of these fans following a LOOP.

It was concluded from the safety evaluation that, when the cable spreading room exceeds 104 degrees F, the effect on cable ampacity is minimal due to the short duration of the ambient temperature rise until cooling is re-established by manual start of the fans. Furthermore, there were no operability concerns identified relating to the capability to maintain acceptable temperatures in the area after the fans are manually started.

2.6.4.11
PER 292-0879

This PER documented a situation where, during a review for subcompartment pressurization of Emergency Core Cooling System (ECCS) pump rooms, it was determined that some penetrations in the common walls between the pump rooms were found to be sealed with silicone foam. The silicone foam was not qualified or rated as a watertight seal by the vendor. Following further review, compensatory measures were established until proper penetration

seals could be installed in the pump rooms. These measures included manual detection and mitigation of line-crack resultant floods (Operator Tours), and blocking open the water-tight door between the High Pressure Core Spray (HPCS) and Control Rod Drive (CRD) pump rooms.

It was concluded from the safety evaluation that the actions to manually detect and mitigate a flooding event are mitigating actions which would neither cause nor increase the probability of occurrence of an accident as previously evaluated in the LBD. Furthermore, the actions taken of change from automatic mitigation or passive barriers to manual detection and mitigation actions would not increase the consequences of an accident previously evaluated.

2.6.4.12
<u>PER 292-0984</u>

This PER documented a situation where it was determined that alternate installed pressure gauges for the Residual Heat Removal (RHR) System pumps and the High Pressure Core Spray (HPCS) System pump did not meet the intent of the ASME Code requirement that the full scale range of each instrument shall be three times the reference value or less. Following further review, the decision was made to use TDAS points for these pumps (TDAS points for these instruments are analog signals with digital output and meet the ASME Code requirements).

It was concluded from the safety evaluation that use of the TDAS instrumentation would provide acceptable accuracy to ensure that the pumps are performing at the required flow and pressure conditions. Therefore, the instrumentation would meet its required design function an no increase in the consequences of an accident previously evaluated would occur.

2.6.4.13
<u>PER 292-1029</u>

This PER documented a situation where it was determined that an annunciator circuit was routed between Division 1 and Division 2 safety-related raceways, which violated the direct bridging criterion of the WNP-2 Design Specification for electrical separation. The circuitry involved pertained to High Pressure Core Spray (HPCS) pump room cooler. Service Water (SW) flow and temperature switches SW-FS-27 and SW-TS-27 respectively. Following further review, compensatory measures in the form of fire tours were established until the design deficiency could be resolved.

It was concluded from the safety evaluation that allowing plant operation with the annunciator bridging circuit in place was acceptable because compensatory measures were established. The compensatory measure was to establish an hourly fire tour, thereby, preventing the spread of any localized fire that may occur post-

accident between redundant divisions. Furthermore, the annunciator circuits were routed in control raceway (low energy circuits) which supply only milliamps of current.

2.6.4.14
PER 292-1191

This PER documented a situation where wetwell airspace temperatures in excess of 140 degrees F, caused by leaking Main Steam Safety Relief Valves (MSRVs), required increased wetwell spray operation and suppression pool cooling. Following further review, it was determined that continued operation could be allowed pending resolution of the problem.

It was concluded from the safety evaluation that there were no credible mechanisms, as a result of higher wetwell air space temperature, which could increase the probability of an accident. Furthermore, design limits for containment analysis were not exceeded, assumptions in the FSAR were not changed, operator action and response time were not altered and the equipment involved was qualified for the operating conditions.

2.6.4.15
PER 292-1222

This PER documented a situation where Auxiliary Steam (AS) System line break mitigation valves AS-V-68A and AS-V-68B were determined to have a total stroke-to-close time of approximately 21 seconds instead of 20 seconds as specified by design. Following further review, it was determined that there was sufficient margin to increase the stroke time limit.

It was concluded from the safety evaluation that increasing valve stroke time extends the accident mitigation time and results in an increase in building profile temperatures of approximately five degrees F. However, the increase in area temperature does not affect the qualification of safety-related equipment required to respond to an accident since the existing qualification envelopes the increase. Furthermore, changing valve closing stroke time would not result in the occurrence of an accident.

2.6.4.16
PER 292-1263

This PER documented a situation where samples of Auxiliary Condensate (CO) System return tank CO-TK-1 were determined to contain tritium activity. The immediate disposition was to continue to monitor tritium levels in the Auxiliary Boiler by periodic sampling, the frequency of which would be determined by trending.

It was concluded from the safety evaluation that operation of the Auxiliary Boiler and associated steam systems as radioactive systems would not alter any assumptions made in previous evaluations of high or moderate energy line breaks, or any other accidents. Therefore, there would be no increase in the probability of occurrence of malfunction of equipment important to safety. In addition, operation of the Auxiliary Boiler and associated steam systems would not be affected by the tritium in the water.

2.6.4.17
<u>PER 292-1338</u>

This PER documented a situation where six fuses on branch circuits of E-PP-US were of a size and/or type other than that specified in design documentation. One of the branch circuits supplied power to Class 2+ equipment required under Regulatory Guide 1.97, "Post Accident Monitoring Instrumentation." The remaining circuitry supplied power to non-safety related, Class 2 loads. The immediate disposition was to allow the fuses to remain installed until they could be replaced with the preferred fuses.

It was concluded from the safety evaluation that failure of these fuses to coordinate under fault conditions would not initiate an accident or event. Furthermore, an evaluation of load information indicated that the as-installed fuse on the branch circuit supplying Class 2+ loads was of sufficient size to perform its safety function, and that faults or failures of the Class 2 branch circuits would not impact the operation of equipment important to safety.

2.6.4.18
<u>PER 292-1359</u>

This PER documented that the WNP-2 ATWS Evaluation did not account for the time required for Standby Liquid Control (SLC) storage tank outlet valves SLC-V-1A and SLC-V-1B to stoke open. The original ATWS analysis did not account for valve opening in the time assumed for boron transport from the SLC system to the reactor.

It was concluded from the safety analysis that the delay in boron transport would increase the time to shutdown the reactor during an ATWS. However, the time delay of approximately 30-to-35 seconds would result in no significant increase in accident consequences. Furthermore, there would be no impact on equipment due to the delay in boron initiation.

2.6.4.19
PER 292-1376

This PER documented a situation where it was determined that 10 fuses were of indeterminate type in instrument rack E-IR-02. The immediate disposition was to allow the fuses to be installed until they could be replaced with the preferred fuses.

It was concluded from the safety evaluation that E-IR-02 is a Class 2 instrument rack of which the branch circuits serve no safety function. Failure of these fuses to properly clear under fault conditions would not initiate an accident or an event. Furthermore, faults or failures on these Class 2 branch circuits would not impact the operation of equipment important to safety and, therefore, would not increase the consequences of an accident.

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 1992.

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 changes that were processed during 1992.

2.6.6.1
Procedure Revision Form for Test Procedure 8.3.227

The procedure for operation of the Residual Heat Removal (RHR) System in the Fuel Pool Cooling (FPC) assist mode was revised by removing steps for recording data and covering of the ventilation ducts in preparation for core offload. In addition, spool pieces were installed and the commitment was made to run RHR/FPC only during shutdown conditions.

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 LBD. Operation of RHR, Loop B, in the FPC assist mode would not cause an accident different from those previously analyzed.

2.6.6.2
Procedure Revision Form for PPM 8.4.70

The procedure for thermal performance testing of room coolers PRA-FC-1A and PRA-FC-1B was revised to enable thermal performance testing of cooling coils in Standby Service Water System pump houses 1A and 1B in response to Generic Letter 89-13. The testing was performed with the respective service water pump in operation to provide cooling water to the coil. To maintain a high pump house temperature during the test, the supply ventilation fan damper setpoints were reset to prevent cold outside air from entering.

It was concluded from the safety evaluation that the ability of the pumps in the pump houses to perform their safety function would not be impacted and, as a result, the consequences of an accident previously evaluated would not be increased. Furthermore, temporary changes in the fan and damper setpoints would not change the existing analysis and, therefore, the probability of an accident was not increased.

2.6.6.3
Procedure Deviation Forms 92-1208, 92-1209 and 92-1464

Procedures for performing channel functional testing of the Loose Parts Monitoring Detection system were changed to allow for continued surveillance testing while Channels 1, 2 and 8 were out of service at various intervals.

It was concluded from the safety evaluation that the Loose Parts Monitoring System is a monitoring system with no accident mitigating features and, therefore, would not impact the consequences of an accident. Furthermore, the loss of monitoring capability in three channels would not create a new type of accident.

2.6.6.4
Procedure Deviation Form 92-1312

The procedure for operation of the Residual Heat Removal System was modified to increase the value below which wetwell temperature must be maintained due to excessive wetwell airspace temperatures caused by leaking Main Steam Relief Valves.

It was concluded from the safety evaluation that there were no credible mechanisms, as the result of higher wetwell airspace temperature, which would increase the probability of an accident. Furthermore, the affected equipment is qualified for the operating conditions and no failures of materials would be anticipated.

2.6.6.5
Procedure Revision Form for PPM 8.3.240

The procedure for motor operated valve differential testing of Residual Heat Removal System (RHR), Loop B, was revised to provide testing instructions for demonstrating that selected motor operated valves would properly operate when required to perform their intended safety function. This testing was being performed in response to Generic Letter 89-10.

It was concluded from the safety evaluation that, since the RHR System is used to mitigate accidents and that the testing would not establish conditions which exceed any safety system design or operating limits, implementation of the procedure would not increase the probability of occurrence of an accident evaluated previously in the LBD. Furthermore, the testing would not create a different type of malfunction than previously evaluated in the LBD.

2.6.6.6
Procedure Deviation Form 92-1106

The procedure for motor operated valve differential pressure testing of the Auxiliary Steam (AS) System was modified to change the required operating mode for testing from Operational Modes 4 and 5, to any mode when heating steam is not required for the reactor building.

It was concluded from the safety evaluation that the test would not change the consequences of a malfunction of equipment important to safety evaluated previously in the LBD. Furthermore, the dynamic stroke testing of the motor operated valves with the AS System operating within system design limits would not create or contribute to a steam line break or introduce any different accidents not evaluated in the LBD.

2.6.6.7
Procedure Revision Form for PPM 8.3.233

This procedure was developed for testing the capability of various Fuel Pool Cooling (FPC) motor operated valves to properly function when subjected to maximum achievable differential pressures. The testing was being performed in response to Generic Letter 89-10.

It was concluded from the safety evaluation that the testing would not increase the consequences of an accident evaluated previously in the LBD. Furthermore, there would be no significant increase in the probability of occurrence of malfunction of equipment important to safety evaluated previously in the LBD.

2.6.6.8
Procedure Revision Form for PPM 9.3.11

This procedure was developed to establish a method for determining reactor core flow such that the result may be used to calibrate associated indications and instrumentation.

It was concluded from the safety evaluation that this activity verifies the proper core flow value and implements the correct calibration of the core flow loop. There would be no effect on the probability of occurrence of an accident evaluated previously in the LBD. Furthermore, the only effect of this activity is to ensure a properly calibrated core flow signal. Accordingly, there would be no effect on the probability of occurrence of malfunction of equipment important to safety.

2.6.6.9
Procedure Revision Form for PPM 4.12.4.7

The procedure which covers unintentional entry into the region of potential core power instabilities was revised to provide for short-term operating strategies for Cycle 8 following the core instability event that occurred on August 15, 1992. Several recommended operating strategies were described for minimizing the occurrence of core instabilities.

It was concluded from the safety evaluation that the recommended strategies would not increase the consequences of an accident previously evaluated in the LBD because they proposed operation further away from core thermal limits previously evaluated as acceptable in the area of instability susceptibility. Operations with increased margin in critical power ratio, radial peaking, axial peaking and core inlet subcooling (coupled with increased surveillance) would not increase the consequences of an accident.

2.6.7 FIRE PROTECTION PROGRAM CHANGES

The following changes involving the Fire Protection Program are reported in accordance with the NRC Letter Dated May 25, 1989 which approved Amendment No. 67 to the Facility Operating License.

2.6.7.1
SCN 88-030, Revision 2

This SCN provided for the relocation of the material in FSAR Section 9.5.1 (Fire Protection System Description) to FSAR Appendix F (Fire Protection Evaluation). The SCN completes the commitment made to the NRC to "complete the FSAR rewrite to clarify and consolidate commitments covered in FSAR Section 9.5.1 and Appendix F." (Reference Letter GO2-87-129, GC Sorensen to JB Martin, dated April 13, 1987.)

The SCN incorporates a number of changes to the fire protection program description which resulted from outstanding SCNs, from previous changes to the fire protection procedures, or from plant design changes.

It was concluded from the 10CFR50.59 Review that the description of the fire protection program as described in the Licensing Basis Documents was modified by this SCN. However, the intent of procedures, processes and commitments in the LBD was not changed, the level of fire protection provided within the plant was not degraded, and the capability to achieve safe post-fire shutdown was not adversely affected by these changes.

2.7 REPORT OF DIESEL GENERATOR FAILURES

This section contains information regarding diesel generator failures, valid and nonvalid, in accordance with the requirements of WNP-2 Technical Specification 4.8.1.1.3. WNP-2 experienced a total of two valid failures and one nonvalid failure in 1992 for the three emergency diesel generator units.

2.7.1 Identity of diesel generator unit and date of failure.

Division One Emergency Diesel Generator (DG-1)
September 20, 1992 (0147 hours).

Number designation of failure in last 100 valid tests:

This was the First Failure of the last 100 valid tests. This Failure was determined to be a "Valid" Failure.

Cause of failure:

During the performance of the Technical Specification required monthly surveillance test, Operations was unable to obtain full load on the Division One Diesel Generator. The unit was unloaded and shut down while a trouble shooting plan was developed.

Corrective measures taken:

The problem was identified to be a governor actuator that would not respond properly to a signal from the governor controls.

The governor actuators used on tandem diesel engines are a "Matched" set and are calibrated at the factory as a set. The actuators on both diesel engines were replaced with a factory calibrated "Matched" set.

Length of time the diesel generator unit was unavailable:

The Diesel Generator was out of service for 109 hours and returned to service at 1445 hours on September 23, 1992.

Current surveillance test interval:

Thirty-one days.

Verification of test interval:

The surveillance test interval of thirty-one days is in conformance with the Technical Specification Requirements and the recommendations of the NRC Regulatory Guide 1.108, position C.2.d.

2.7.2 Identity of diesel generator unit and date of failure:

Division Three Emergency Diesel Generator (DG-3) September 28, 1992 (0200 hours).

Number designation of failure in last 100 valid tests:

This was the First Failure of the last 100 valid tests. This Failure was determined to be a "Valid" Failure.

Cause of failure:

During the performance of the Technical Specification required monthly surveillance test, Operations was unable to maintain full load on the Division Three Diesel Generator. The unit would accept load and when the Operator released the governor control switch, the load would begin decreasing without any Operator action. The unit was unloaded and shut down while a trouble shooting plan was developed.

The problem was identified to be a governor control switch in which a cam internal to the switch had come loose. This allowed the "Lower" side of the control switch to be actuated whenever the Operator would release the control switch handle. This caused the load to be reduced on the generator.

Corrective measures taken:

A new switch was installed and the diesel generator unit tested satisfactorily.

Length of time the diesel generator unit was unavailable:

The Diesel Generator was out of service for 70 1/2 hours and returned to service at 0023 hours on October 1, 1992.

Current surveillance test interval:

Thirty-one days.

Verification of test interval:

The surveillance test interval of thirty-one days is in conformance with the Technical Specification Requirements and the recommendations of the NRC Regulatory Guide 1.108, position C.2.d.

2.7.3 Identity of diesel generator unit and date of failure:

Division Two Emergency Diesel Generator (DG-2) November 30, 1992 (0914 hours).

Number designation of failure in last 100 valid tests:

This was the First Failure of the last 100 valid tests. This test was determined to be a nonvalid failure.

Cause of failure:

During the performance of the Technical Specification required monthly surveillance test, the Division Two Diesel Generator failed to obtain nominal voltage within the required ten seconds. The unit did obtain rated voltage in approximately 16 seconds without any Operator intervention.

Corrective measures taken:

The normal troubleshooting procedures were implemented, which included subsequent test runs (three fast starts). The event did not repeat itself during the testing. A decision was made to replace the voltage regulator because it was found out of calibration. All other apparent causes for the failure to achieve the required voltage were eliminated. The Diesel Generator retested satisfactorily and was returned to service.

Length of time the diesel generator unit was unavailable:

The Diesel Generator was out of service for 87 hours and returned to service at 0021 hours on December 4, 1992. A waiver of compliance was requested and granted from the NRC for an extension of the action statement.

Current surveillance test interval:

Thirty-one days.

Verification of test interval:

The surveillance test interval of thirty-one days is in conformance with the Technical Specification Requirements and the recommendations of the NRC Regulatory Guide 1.108, position C.2.d.