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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • 3000 George Washington Way • Richland, Washington 99352

Docket No. 50-397

February 28, 1990

Mr. J. B. Martin
Regional Administrator
Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane, Suite 210
Walnut Creek, California 94596

Dear Mr. Martin:

Subject: NUCLEAR PLANT NO. 2 ANNUAL REPORT

Reference:
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 Operating Information Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Report for calendar year 1989. Should you have any questions or comments please contact G. L. Gelhaus, WNP-2 Assistant Plant Technical Manager.

Very truly yours,

JW Baker

C. M. Powers
WNP-2 Plant Manager

/bc
Attachments

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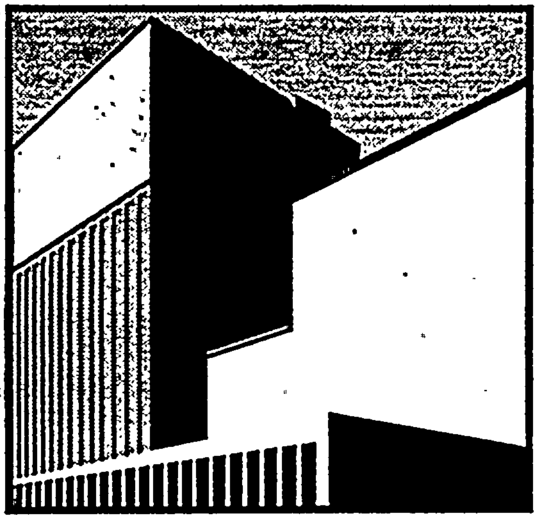
WNP-2 ANNUAL OPERATING REPORT 1989



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM

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TABLE OF CONTENTS

ANNUAL OPERATING REPORT

OF

WNP-2

FOR 1989

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

1.0 INTRODUCTION

The 1989 Annual Operating Report of Washington Public Power Supply System Plant Number 2 (WNP-2) is provided as a supplement to the Monthly Operation Report. This report is submitted in accordance with the requirements of Federal Regulations and Facility Operating License NPF-21. It should be noted that, for ease of reference and completeness, additional reports are also included. Plant WNP-2 is a 3323 MWt, BWR-5, which began commercial operation on December 13, 1984.

On January 30, 1989 the reactor scrammed due to Turbine Control Valve Fast Closure actuation of the Reactor Protection System (RPS) logic. The RPS logic was actuated when the main generator 500Kv output breakers tripped as a result of high currents created when a porcelain insulator, on the output side of the 25/500Kv main transformer, shorted to ground. On February 2, 1989, following a three-day outage to replace and clean several insulators, the Plant returned to normal operation. However, due to a problem with opening one of the four outboard Main Steam Isolation Valves, power output was limited to 78 percent. The Plant continued to run in this configuration, with the valve closed and steam supplied through three of four main steam lines, until the Plant was shut down for the annual maintenance and refueling outage.

From April 28, 1989 until July 2, 1989 the Plant was in a shutdown condition as scheduled for the annual maintenance and refueling outage. Following the outage, the Plant was restarted and operated until August 6, 1989 when a reactor scram occurred due to the trip of a reactor feedwater pump caused by a problem in the feedwater pump control oil system. On August 9, 1989 the Plant was restarted. The Plant was shut down and an Unusual Event was declared on August 11, 1989 as a result of declaring six Class 1E, 480 volt A.C., Motor Control Centers (MCCs) inoperable due to the discovery of a design deficiency. All of the affected MCC power supply circuit breakers were replaced with jumper cables of equal capacity with the exception of one, which was replaced with a fused disconnect of equal capacity. During restart, on August 17, 1989, another reactor scram occurred as a result of a surveillance being performed on a reactor level instrument associated with the Automatic Depressurization System (ADS).

On August 18, 1989 the reactor was restarted but power output was limited to 70 percent due to removing a reactor feedwater pump from service because of a bearing failure. One of two reactor pumps was repaired and the Plant returned to full power on September 13, 1989. The Plant essentially remained at 100 percent power until September 21, 1989, when it was shut down due to two leaking condenser tubes and two ruptured bellows connections on a low-pressure steam extraction line. Repairs were made and the Plant was restarted on September 29, 1989 and ran at or near 100% capacity for the remainder of the year (94 days).

During 1989, 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.

(a) The fourth refueling outage was successfully completed. Significant activities included:

- o Preventive maintenance on the eight Main Steam Isolation Valves (MSIVs), and a major overhaul on the valve that limited Plant power output to 78 percent of capacity. Four of the valves were repaired to reduce the potential for valve binding caused by galling in the cylinders. Valve pistons of a new design by Rockwell, the MSIV supplier, were installed in those MSIVs.

- o Overhaul of one of the two Reactor Feedwater Pump Turbines. The turbine was dismantled and the rotors were cleaned and inspected for cracks or other defects.

- o Inspection of two of the three Diesel Generators. This task included replacing power assemblies in the two engines.

- o Inspection of one of three Low-Pressure Turbine Rotors. Non-destructive examination of the rotor confirmed 17 crack indications and the blades were replaced. Subsequent evaluations determined that the problems were limited to this single rotor.

- o Maintenance on 40 Control Rod Drive Mechanisms (CRDMs). This activity included removing, replacing and rebuilding the CRDMs.

- o Removal of a radioactive "hot spot" in the vessel drain to the Reactor Water Cleanup System. This activity required the installation of a temporary bottom head drain plug in the Reactor Pressure Vessel. The plug was installed from the top of the vessel which required removal of four fuel assemblies, a control rod blade, guide tube and associated support pieces.

- o Removal of spent fuel assemblies and refueling the reactor. The refueling activity included replacing 136 fuel assemblies, using a fuel shuffle scheme.

(b) In terms of electrical output, WNP-2 delivered 6.1 billion kilowatt-hours to the Bonneville Power Administration, surpassing the previous year's record by more than 117 million kilowatt-hours. In addition, the capacity factor for 1989 was a Plant record 63.78 percent (up from 62.38 percent in 1988).

(c) A new monthly generation mark was established in December, when 780 million kilowatt-hours were generated.

(d) In December, WNP-2 celebrated five years of commercial operation. Since 1985, the Plant has provided more than 28 million megawatt-hours of electricity to the Bonneville Power Administration.

In 1989, total radiation exposure at the Plant was 492 man-rem, as compared to the 1988 level of 352 man-rem. (The Institute for Nuclear Power Operation (INPO) has set 460 man-rem as the 1990 industry goal for BWRs.) Contributing to this increase were the following activities:

- Removal and replacement of 40 Control Rod Drive Mechanisms (CRDMs). During work on the first 20 CRDMs, the man-rem exposure was 13.005. As a result of that exposure, temporary shielding was put into place and the man-rem exposure was reduced to 5.635 for the remaining 20 CRDMs.

- Removal of the "hot spot" in a reactor bottom head drain line elbow. Before replacement, the "hot spot" area was reading between 2,000 and 3,000R. Total man-rem exposure for this activity was 18.836.

- Modifications to the Control Rod Drive Rebuild Room and flushing of Low-Pressure Core Spray (LPCS) System lines. These actions were taken to reduce man-rem exposure in the future.

During the year WNP-2 received 23 Notices of Violation (NOVs): One (1) Level III, twenty-one (21) Level IV and one (1) Level V. The Level III violation was associated with commercial grade dedication issues and included a proposed $50,000 civil penalty.

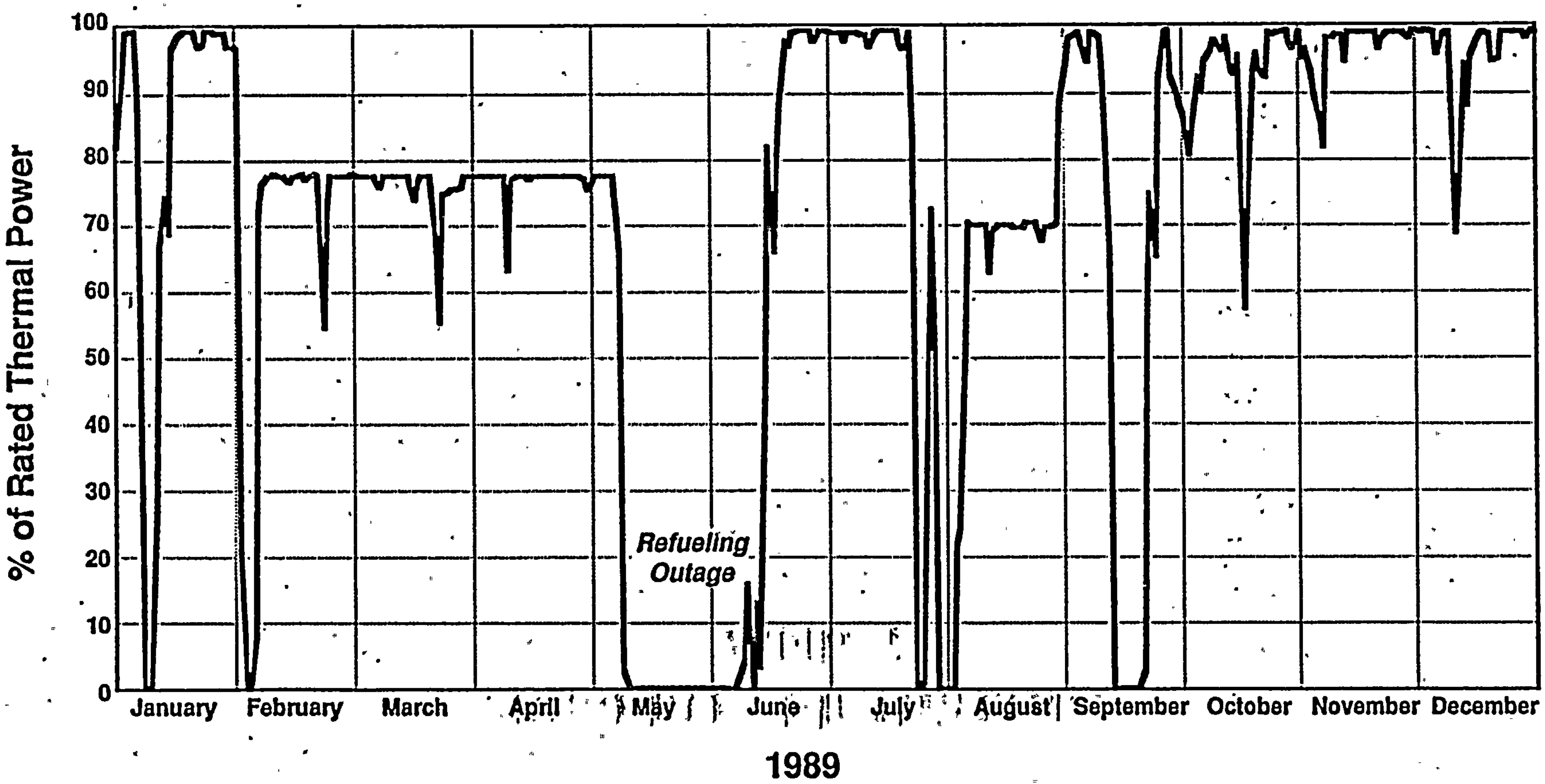
Also during 1989, a total of 45 Licensee Event Reports (LERs) were written and submitted pursuant to the requirements of 10CFR50.73.

The 1989 capacity factors, based upon net electrical energy output, are listed in the following table.

Month	Capacity Factor
January	76.17
February	68.25
March	73.02
April *	68.32
May	0
June **	0
July	88.47
August	52.94
September	58.39
October	88.80
November	95.84
December	95.74
Overall	63.78

* Started Maintenance/Refueling Outage
** Ended Maintenance/Refueling Outage

WNP-2
1989 POWER HISTORY



Data based on average power generated per day. Therefore, recovery from a scram that ocurred within a 24 hour period will not indicate a zero percent power level.

900023

2.0 REPORTS

The reports provided in this section meet the requirements of Federal Regulations (10CFR50.59) and the WNP-2 Operating License. Complete data for the year 1989 has been included.

2.1 ANNUAL PERSONNEL EXPOSURE AND MONITORING REPORT

RER-020

WASHINGTON [illegible]IC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT / 1.16 APPENDIX A

01/26/90 11:[illegible]

NUCLEAR PLANT NO. 2

REPORT FOR CALENDAR YEAR 1989

		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	68.105	0.073	55.753	40.102	0.027	31.527
	OPERATING PERSONNEL	42.537	0.000	0.000	32.471	0.000	0.000
	HEALTH PHYSICS PERSONNEL	26.767	0.037	15.620	22.130	0.004	8.943
	SUPERVISORY PERSONNEL	15.073	1.816	0.283	6.090	1.472	0.089
	ENGINEERING PERSONNEL	14.164	9.165	13.565	4.036	4.399	6.090
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	23.592	0.139	23.785	14.824	0.054	11.054
	OPERATING PERSONNEL	1.916	0.000	0.000	1.571	0.000	0.000
	HEALTH PHYSICS PERSONNEL	7.561	0.000	1.266	3.803	0.000	1.213
	SUPERVISORY PERSONNEL	2.368	0.000	0.330	0.980	0.000	0.103
	ENGINEERING PERSONNEL	3.619	2.498	4.002	1.052	0.884	0.669
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	2.790	0.000	7.869	1.799	0.000	3.919
	OPERATING PERSONNEL	2.101	0.000	0.000	1.512	0.000	0.000
	HEALTH PHYSICS PERSONNEL	1.142	0.000	0.589	1.597	0.000	0.622
	SUPERVISORY PERSONNEL	0.715	0.184	0.010	0.487	0.179	0.004
	ENGINEERING PERSONNEL	3.313	3.116	9.977	1.222	1.120	2.316
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	116.436	1.776	161.313	89.675	0.747	99.971
	OPERATING PERSONNEL	1.493	0.000	0.000	0.955	0.000	0.000
	HEALTH PHYSICS PERSONNEL	4.207	0.000	30.639	6.723	0.000	30.502
	SUPERVISORY PERSONNEL	3.313	0.000	3.386	2.484	0.000	1.857
	ENGINEERING PERSONNEL	10.462	6.805	13.340	3.848	2.766	5.127
WASTE PROCESSING	MAINTENANCE PERSONNEL	8.455	0.000	0.000	5.319	0.000	0.000
	OPERATING PERSONNEL	0.070	0.000	0.000	0.022	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.467	0.000	2.700	0.523	0.000	2.797
	SUPERVISORY PERSONNEL	0.000	0.000	0.991	0.000	0.000	1.902
	ENGINEERING PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
REFUELING	MAINTENANCE PERSONNEL	15.235	0.013	1.220	13.147	0.004	0.761
	OPERATING PERSONNEL	1.009	0.000	0.000	0.714	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.528	0.000	2.596	0.723	0.000	1.257
	SUPERVISORY PERSONNEL	0.542	0.000	0.000	0.210	0.000	0.000
	ENGINEERING PERSONNEL	1.336	0.604	0.220	0.358	0.206	0.067
TOTAL	MAINTENANCE PERSONNEL	234.613	2.001	249.940	164.866	0.832	147.232
	OPERATING PERSONNEL	49.126	0.000	0.000	37.245	0.000	0.000
	HEALTH PHYSICS PERSONNEL	40.672	0.037	53.410	35.499	0.004	45.334
	SUPERVISORY PERSONNEL	22.011	2.000	5.000	10.251	1.651	3.955
	ENGINEERING PERSONNEL	32.894	22.188	41.104	10.516	9.375	14.269
GRAND TOTAL		379.316	26.226	349.454	258.377	11.862	210.790

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information concerning main steam line safety/relief valve challenges for calendar year 1989 in accordance with the requirements of NUREG 0737, Item II.K.3.3, and as required by WNP-2 Technical Specifications, Administrative Controls section, paragraph 6.9.1.5(b).

First Quarter

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
01/30/89	MS-RV-1B	A	E	B	6%	89-002
01/30/89	MS-RV-1C	A	E	B	6%	89-002

The January 30, 1989 actuations were in response to a turbine trip - reactor scram transient.

Second Quarter

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
04/29/89	MS-RV-1A	B	D	C	17.8%	—
04/29/89	MS-RV-2A	B	D	C	17.8%	—
04/29/89	MS-RV-3A	B	D	C	17.8%	—
04/29/89	MS-RV-4A	B	D	C	17.8%	—
04/29/89	MS-RV-1B	B	D	C	17.8%	—
04/29/89	MS-RV-2B	B	D	C	17.8%	—
04/29/89	MS-RV-3B	B	D	C	17.8%	—
04/29/89	MS-RV-4B	B	D	C	17.8%	—
04/29/89	MS-RV-5B	B	D	C	17.8%	—
04/29/89	MS-RV-1C	B	D	C	17.8%	—
04/29/89	MS-RV-2C	B	D	C	17.8%	—
04/29/89	MS-RV-3C	B	D	C	17.8%	—
04/29/89	MS-RV-4C	B	D	C	17.8%	—
04/29/89	MS-RV-5C	B	D	C	17.8%	—
04/29/89	MS-RV-1D	B	D	C	17.8%	—
04/29/89	MS-RV-2D	B	D	C	17.8%	—
04/29/89	MS-RV-3D	B	D	C	17.8%	—
04/29/89	MS-RV-4D	B	D	C	17.8%	—
04/29/89	MS-RV-1C	C	D	C	0%	—

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

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
04/29/89	MS-RV-4B	C	D	C	0%	--
04/29/89	MS-RV-2B	C	D	C	0%	--
04/29/89	MS-RV-3B	C	D	C	0%	--
04/29/89	MS-RV-5B	C	D	C	0%	--
04/29/89	MS-RV-2C	C	C	C	0%	--
04/29/89	MS-RV-1B	C	D	C	0%	--

The April 29, 1989 manual actuations were in response to valves being cycled to test acoustic monitors. The April 29, 1989 spring actuations were in response to the valves being "simmered" four times for in-situ setpoint verification testing.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
06/26/89	MS-RV-3A	C	C	C	1.5%	--
06/26/89	MS-RV-1B	C	C	C	1.5%	--

The June 26, 1989 actuations were in response to the valves being "simmered" two times for in-situ setpoint verification testing.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
06/27/89	MS-RV-4A	C	C	C	1.5%	--
06/27/89	MS-RV-2B	C	C	C	1.5%	--

The June 27, 1989 actuations were in response to the valves being "simmered" two times for in-situ setpoint verification testing.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
06/28/89	MS-RV-1B	B	C	C	13.0%	--
06/28/89	MS-RV-4D	B	C	C	13.5%	--
06/28/89	MS-RV-5C	B	C	C	13.5%	--
06/28/89	MS-RV-4A	B	C	C	13.5%	--
06/28/89	MS-RV-3D	B	C	C	13.5%	--
06/28/89	MS-RV-4B	B	C	C	13.5%	--
06/28/89	MS-RV-4C	B	C	C	13.5%	--
06/28/89	MS-RV-5B	B	C	C	13.5%	--
06/28/89	MS-RV-1B	B	C	C	13.5%	--
06/28/89	MS-RV-2C	B	C	C	13.5%	--
06/28/89	MS-RV-2A	B	C	C	13.5%	--

The June 28, 1989 actuation of MS-RV-1B was in response to the valve being manually actuated to clear seats and reseat to reduce leakage. The remainder of the June 28, 1989 actuations were in response to the valves being cycled to verify operability and to test the acoustic monitors.

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

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
08/16/89	MS-RV-5C	B	C	C	10%	--

The August 16, 1989 actuation was in response to the valve being cycled to test the acoustic monitor.

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

NOTES: 1) Remote manual actuations occurred in support of acoustic monitor position indication calibration testing required by Technical Specifications LCO 3/4.4.2.

2) Spring set testing was performed in accordance with ASME Section XI and Technical Specifications requirement in applicability paragraph 4.0.5.

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
1/7/89	S	82.9	A	1	--	HC	HTEXCH-D	The Plant was shut down to correct a condenser tube in-leakage problem. Repairs were performed and unit returned to service.
1/30/89	F	73.2	A	3	89-002	EB	ELECON	The generator tripped at 100% power due to a fault on an insulator between the main step-up transformers and generator disconnects. The insulator was replaced and inspection/cleaning was performed on remaining insulators prior to startup.
2/2/89	F	10.4	H	1	--	HF	ZZZZ	Generator was removed from service due to main condenser vacuum problems. Low vacuum was caused by high condensate temperature as a result of cooling tower problems.
2/18/89	S	0	H	5	--	RB	CONROD	Reduced power to perform a scheduled control rod sequence exchange.
3/17/89	F	0	H	5	89-006	AA	ZZZZZ	Reduced load due to an engineering analysis which indicated post loca potential "integrated dose rate" to control room personnel through ventilation system would exceed Tech Spec Limitations. After additional evaluation, errors were discovered in calibration methodology which alleviated the finding of the previous analysis.

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS (Continued)

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
4/28/89 thru 6/28/89	S	1456.8	C	1	--	RC	FUELXX	The plant was shut down as scheduled for the annual refueling and maintenance outage.
6/29/89	S	33.5	B	3	89-028	HA	MECFUN	Generator was removed from grid to perform overspeed tests on turbine. A reactor scram occurred prior to completion of tests.
6/30/89	S	31.7	B	1	--	HA	MECFUN	Generator was removed from grid to complete overspeed testing of turbine and perform scram time testing.
8/6/89	F	59.6	A	3	89-031	CH	TURBIN	Reactor scram from 100% power on Low RPV Level. Initiated by trip of "B" reactor feedwater drive turbine on low lube oil pressure during testing of backup oil pumps.
8/11/89	F	115	F	1	89-034	EB	ELECON	Plant was shut down to resolve and correct electrical fuse coordination and separation issues on safety-related low voltage motor control centers.
8/17/89	F	17.3	G	3	89-035	IA	INSTRU	Reactor scram from 67% power due to inadvertent actuation of an RPV Low Level switch during execution of a Tech Spec Surveillance.

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS (Continued)

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
9/21/89	F	29	A	1	--	HC	HTEXCH-D	Plant was manually shut down because of rapidly increasing conductivity due to condenser tube in-leakage. Two damaged tubes were plugged and plant remained down for repair of steam extraction lines.
9/22/89	S	167.2	A	4	--	HJ	PIPEXX-E	Plant remained down for repair/ replacement of two failed steam extraction line expansion bellows and condenser baffle repair. After completion of repair to all damaged components, the plant was returned to service.
10/23/89	S	0	H	5	--	RB	CONROD	Reduced power to perform a scheduled control rod sequence exchange.
12/12/89	S	0	H	5	--	RB	CONROD	Reduced power to perform a control rod sequence exchange.

CAUSE CODE	TOTAL FOR 1989	TOTAL GENERATOR OFF-LINE HOURS
A	5	411.9
B	2	65.2
C	1	1456.8
D	0	0
F	1	115.0
G	1	17.3
H	5	10.4
	TOTAL	2076.6

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS (Continued)

SUMMARY OF CODES

OUTAGE TYPE	CAUSE CODE	SHUTDOWN METHOD	SYSTEM CODE	SYSTEM DESCRIPTION
F - Forced	A - Equipment Failure	1 - Manual	AA	Air Conditioning, Heating, Cooling & Ventilation Controls
S - Scheduled	B - Maintenance or Test	2 - Manual Scram	CH	Feedwater Systems & Controls
	C - Refueling	3 - Auto Scram	EB	AC Onsite Power Systems & Controls
	D - Regulatory Restriction	4 - Continued	HA	Turbine Generator & Controls
	E - External Cause	5 - Reduced Load	HC	Main Condenser Systems & Controls
	F - Administration	9 - Other	HF	Circulating Water Systems & Controls
	G - Personnel Error		HJ	Other Features of Steam & Power Conversion Systems (not included elsewhere)
	H - Other		IA	Reactor Trip Systems
			RB	Reactivity Control Systems
			RC	Reactor Core

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS (Continued)

SUMMARY OF COMPONENT CODES

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Control Rod Drive Mechanism (CONROD)	Control Rod Drive Mechanism
Electrical Conductors (ELECON)	Bus Cable Wire
Fuel Elements (FUELXX)	
Heat Exchangers (HTEXCH)	Condensers Coolers Evaporators Regenerative Heat Exchangers Steam Generators Fan Coil Units
Instrumentation and Controls (INSTRU)	Controllers Sensors/Detectors/Elements Indicators Differentials Integrators (Totalizers) Power Supplies Recorders Switches Transmitters Computation Modules
Mechanical Function Units (MECFUN)	Mechanical Controllers Governors Gear Boxes Varidrives Couplings

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Pipes, Fittings (PIPEXX)	Pipes Fittings
Turbines (TURBIN)	Steam Turbines Gas Turbines Hydro Turbines
Codes Not Applicable (ZZZZZZ)	

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
E-C1-1	DC Power System Battery Charger	During performance of weekly surveillance, Division I 125 Volt battery charger would not develop required voltage.	Silicon Control Rectifier (SCR) firing board was found defective.	Replaced SCR firing board with new like board. Performed operability test and verified charger operated properly.
E-C1-2	DC Power System Battery Charger	While working in the area, an electrician noticed that the charger was making an unusual noise (with the Plant in normal operation).	Defective silicone card. Rectifier caused damage to firing board.	Replaced damaged firing board with same. Replaced defective silicone card rectifier with same.
DSA-C-1C	HPCS Power - Diesel Starting Air (DSA)	While performing preventive maintenance on DSA High Pressure Core Spray diesel generator during Plant operation, it was noted that the unloader valve fitting and discharge valve were damaged.	Previous installation or repair had apparently overtightened valve and fitting resulting in cracking.	Replaced unloader valve and discharge valve with same.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM, DESCRIPTION	CAUSE	ACTION TAKEN
DSA-C-2C	HPCS Power - Diesel Starting Air (DSA)	While on tour, an operator noticed air leaking from the air intake on the backup starting air compressor for the HPCS diesel generator DG-1C.	Condensation in the line from the after cooler caused rust and corrosion on the valve seats, which created a leak path.	Replaced the suction and discharge valves with same type. Verified compressor operation.
CRD-CB-P1A	Control Rod Drive	With the reactor in Mode 1 at a 100 percent power, the racking mechanism was found broke for the 4160 Volt circuit breaker. This was discovered while attempting to return the Control Rod Drive "A" to service after repairing a minor seal leak.	Racking mechanism broke probably due to wearout.	Replaced broken racking mechanism with same type new racking mechanism. Torqued hold-down bolts to 41 ft/lbf. Observed racking in and verified pump motor operated.
RCIC-42-S21A5C	Reactor Core Isolation Cooling	While trouble shooting the failure of the RCIC minimum flow return valve to the Suppression Pool to open or to close, an undervoltage relay in the valve motor starter was found burned up. The problem was identified during MOVATS testing.	Cause unknown; probably wearout.	Replaced undervoltage relay. Tested satisfactorily.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RPS-EPA-3D	Reactor Protection System	During performance of surveillance channel functional test, a failure occurred in the undervoltage trip functional check for the Reactor Protection System electrical protection assembly. The undervoltage trip has a setpoint of 110.7 to 108.5 Volts and an allowable value of 108 Volts. The undervoltage trip occurred at 50.1 volts.	Cause of failure traced to defective circuit board. Cause of defective circuit board is unknown.	Replaced the circuit board with the same type. Modification request issued to request new design of circuit boards to be implemented at next scheduled outage.
E-IN-2	Instrument AC Power Supply Inverters	An attempt to switch critical instrument power inverter showed inverter would not synchronize or transfer forward.	Synchronize input not being transmitted to logic due to dirty contacts on K7 relay.	Burnished contacts and functionally tested unit.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
E-IN-3	Instrument AC Power Supply Inverters	Fuses to Division I critical instrument Power blew resulted in tripping inverter unit to alternate power source and giving inverter trouble alarm. Fuses have blown repeatedly.	Static switch logic board (piece part of the inverter) was found conducting one-half of load amps from the inverter and one-half from alternate power supply due to faulty static switch gate firing module. This failure was due to wearout and aging.	Replaced static switch gate firing module with same and tested functional.
SGT-ESH-1A	Standby Gas Treatment	The electric strip heater bank 2A on the "A" Standby Gas Treatment train low temperature alarm would not clear. Annunciation of temperature is in the Main Control Room.	The heater wire where it connects to the heaters had frayed insulation causing the wire to short against the heater box cover.	Replaced shorted heaters with spare heaters. Taped frayed insulation test satisfactorily.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RHR-P-2B	Residual Heat Removal - Low Pressure Injection	Information was received that the Residual Heat Removal (RHR) pumps at LIMERIC experienced loosening of pump hold down bolts. WNP-2 found one RHR pump which had hold down bolts that did not meet specifications.	Cause of loose bolts on the RHR pump was attributed to probable thermal cycling causing the bolts to relax.	Retorqued the bolts on RHR pump No. 2 to specification. Initiated preventive maintenance measures to verify adequate torquing on hold down bolts on all emergency core cooling systems.
RRC-P-1A	Reactor Recirculation	An attempt to run Reactor Recirculation pump "A" at 60 Hertz was unsuccessful. The pump was running at 15 Hertz at the time.	The position switch on the flow control valve permissive input to pump 60 Hertz logic was not making up with valve in closed position.	Inserted spacers between limit switch mounting bracket and limit switches. Replaced mounting bolts with longer bolts to provide sufficient thread engagement to prevent bracket loosening. Performed voltage check to verify proper operability.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-RLY-K81D	Main Steam	Manual isolation relay exhibited excessive noise when running.	Normal wear expected in GE "HFA" relays.	Replaced with rebuilt relay. Removed relay to be rebuilt and reused.
RCIC-DT-1	Reactor Core Isolation Cooling	Check of suspected leakage on inboard gland seal showed leakage in the interface area between turbine casing and gland seal upper housing.	Carbon glands degraded. Sealant between gland seal housing and casing halves showed wear and deterioration.	Replaced carbon glands. Removed old sealant and applied tempflex joining compound. Performed in-service leak test.
RFW-DT-1B	Reactor Feedwater	During unrelated maintenance on Reactor Feedwater Pump "B", oil was observed leaking in the vicinity of the hydraulic trip assembly.	Infra-red examination of hydraulic trip assembly piping showed leak through of the in-line check valve probably due to wearout.	The check valve was replaced with the same type. Visual inspection of the hydraulic trip assembly was performed to verify no leakage.

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
CRD-FCV-2A	Control Rod Drive	During routine observation, the indicated flow through the Control Rod Drive flow controller valve 2A was observed to be oscillating between 5 and 10 gpm when the controller was in automatic.	The reason for flow oscillations was attributed to a faulty valve positioner. The cause of faulty valve positioner was normal wear.	Replaced faulty valve positioner with spare. Recalibrated valve and tested satisfactorily.
MS-V-28A	Main Steam	Main Steam Isolation Valve 28A failed local leak rate test.	Leak by main body seat due to surface imperfections.	The main body seat was machined. The valve was retested and leakage was acceptable.
PSR-V-X77A/1 PSR-V-X77A/2	Post-Accident Sampling Radioactive	Containment isolation valve for post-accident sampling system failed local leak rate testing.	Cause unknown. Probably due to valve disc wear.	Valve was disassembled and inspected. Mating surfaces were lapped to remove minor rough spots and burrs. Valve was reassembled and leak tested satisfactorily.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RFW-V-10B	Reactor Feedwater	During local leak rate surveillance testing, the reactor feedwater swing check valve 10B would not seal.	The Stillman EP soft seat seal had excessive wear.	Replaced the Stillman EP soft seat seal. Established four year equipment qualification schedule for soft seal replacement.
RFW-V-65A	Reactor Feedwater	During performance of local leak rate testing, Reactor Feedwater supply isolation valve showed excessive leakage.	Valve seat and disc scratched due to unknown causes.	Valve seat and disc lapped. Valve repacked and torqued. Local leak rate test performed satisfactorily.
SW-V-165B	Standby Service Water	Normal observation found the 18-inch spray pond "B" ring by-pass valve leaking by the seat.	Normal seat wear.	Replaced seat and thrust collar. Valve tested satisfactorily.
SW-V-214	Standby Service Water	During surveillance testing of "A" emergency diesel generator, engine tripped on high temperature.	Standby Service Water valve which supplies cooling water to diesel engine heat exchanger failed to open due to disc separating from stem, most likely due to wearout.	Valve was temporarily removed from service and replaced with a spool piece until a new valve can be purchased and installed.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
SW-V-220A	Standby Service Water	Attempted to operate Standby Service Water makeup valve to "A" diesel engine cooling water. Valve would not operate.	Broken stem nut and lock nut resulted in stem assembly failure and valve could not be operated.	Replaced stem nut and lock nut. Valve retested satisfactorily.
HPCS-MO-15	High Pressure Core Spray	During surveillance testing, Suppression Pool suction valve would not operate via the motor. Motor ran, but valve would not move.	The worm shaft clutch gear assembly fell apart due to missing split spacer which acts as a seat for shaft set screws. The spacer was not installed during manufacturing.	Replaced entire clutch assembly and worm shaft. Revised procedures to inspect assembly and re-stake set screws if required. Notified Limitorque of possible quality control deficiencies (10CFR Part 21 Report). Verified operability of all safety related valves with Limitorque motor operators of similar design during refueling outage.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
LPCS-MO-12	Low Pressure Core Spray	During performance of the annual stroke times surveillance on the Low Pressure Core Spray System, valve operator #12 indicated closed with 3500 gpm through the valve.	Limit switches on the valve operator, Rotor 2, were found out of adjustment. Cause of adjustment problem was unknown.	Limit switches were adjusted to close at 5% open. Valve functionally tested satisfactorily.
RCIC-MO-8	Reactor Core Isolation Cooling	Reactor Core Isolation Cooling turbine steam supply valve trips overloads when operated.	Limit switch #16 found closed and rotor locked due to attempting to stroke valve closed when it was 98% closed.	Adjusted limit switch and verified it opened during valve closing.
RFW-MO-112B	Reactor Feedwater	During surveillance test, motor operator for Reactor Feedwater high pressure heater 6B outlet isolation valve blew all three line fuses during open stroke. Valve remained in partially open position. The Plant was operating at 71% power.	Rotors 1 and 2 on valve position limit switches were out of adjustment.	Adjusted rotors to actuate as required to properly open and close valve. The valve was retested satisfactorily.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
SW-MO-2A	Standby Service Water	After routine start of Standby Service Water "A" pump, it was noted that outlet valve was indicating intermediate open position. Valve was found 30 percent open. Operator on tour was directed to manually open valve to 100 percent.	Valve operator had broken worm gear and associated gear damage due to essentially no grease in gear case.	Valve operator worm gear and associated gearing was repaired. Valve was returned to service and functionally tested. Five other valves were inspected in the Standby Service Water System to verify proper lubrication.
SW-MO-4B	Standby Service Water	During surveillance testing, Standby Service Water inlet valve to diesel generator #2 blew a fuse in one of the phase lines during the close cycle.	Rotor #1 did not move when valve closed.	Adjusted number 1 rotor to open with handwheel 3.5 turns off of seated position. Tested for proper operability.
SW-MO-44	Standby Service Water	While attempting to operate Standby Service Water supply valve to Low Pressure Core Spray room cooler, valve would not operate electrically and smoke was observed at the valve operator.	Motor leads T1 and T3 were nicked by improper installation of motor cover.	The wires were spliced and T1 was relugged.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
SGT-ESH-1A SGT-ESH-2A	Standby Gas Treatment	The electric strip heater low temperature alarms would not clear. Annunciation of temperature is in the Main Control Room.	The heater wire where it connects to the heaters had frayed insulation causing the wire to short against the heater box cover.	Replaced shorted heaters with spare heaters. Taped frayed insulation test satisfactorily.
DCW-TS-12A1	Diesel Cooling Water	During the Spring refueling outage, while testing the 1A Emergency Diesel Generator, the Diesel Cooling Water Temperature Sensing Element was found leaking.	Unknown-Probably wearout.	Replaced and tested temperature sensing switch.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
LD-TS-619B	Leak Detection	With the plant operating routine observation found the output relay chattering on temperature trip unit 619B of the leak detection system on the main steam line to the turbine building. Contacts on the temperature switch were worn due to normal wear.	Replaced the temperature trip unit 619B on the leak detection system on the main steam line to the turbine building.	
MS-LIS-24B MS-LIS-24D	Main Steam System	During performance of the monthly surveillance test in the normal operating mode the level indicating switches (LIS) would not trip as required. These switches provide input to the Reactor Protection System.	Contacts on the switch were worn due to normal wear.	Changed wiring to use a spare switch. Performed surveillance testing.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-LIS-31C	Main Steam System	During the Spring Refueling Outage while performing surveillance testing on Reactor Low Level 2 it was noted that the level indicating switch (LIS) had considerable bounce.	These switches have a history of failure due to age.	Replaced switch with the same type.
MS-LIS-37A	Main Steam System	During the Spring Refueling Outage while filling the reactor vessel level trip unit exhibited considerable bounce.	The microswitch in the Level Indicating Switch (LIS) was not functioning properly causing excessive movement. Probable cause is normal wear.	Replaced and tested the micro switch.
MS-RIS-610D	Radiation Monitoring	During the performance of routine surveillance testing with the plant at power the main steam (MS) line radiation monitor (RIS) channel "D" downscale and Reactor Protection System trips would not clear.	The switch would not reset due to normal aging and cyclic fatigue.	Removed defective drawer and replaced with the same type spare drawer. Surveillance test performed.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RHR-PS-16A	Residual Heat Removal	With the plant operating normal observation revealed the Residual Heat Removal (RHR) Pressure Switch (PS) which is an Automatic Depressurization System (ADS) permissive failed.	Cause unknown- probably wearout.	Replaced pressure switch with like kind and performed test.
SGT-TS-2A11	Standby Gas Treatment	With the plant at power, the annunciator for a carbon adsorber strip heater low temperature on the standby gas treatment system came on and stayed in. The strip heater would not heat.	Contacts for the temperature switch (TS) showed open at ambient temperature. Cause most likely due to age or wearout.	Replaced temperature switch and performed a satisfactory test.
SLC-LS-600	Standby Liquid Control	While performing a surveillance test, with the plant operating at power, the Standby Liquid Control (SLC) tank level indicating meter was sticking at mid scale.	Cause is unknown. Probably wearout of Level Switch (LS).	Replaced level indicator and level switch. Completed surveillance test.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
SW-PS-1A	Standby Service Water	During Spring Refueling Outage, while performing preventative maintenance on the Service Water Pump the pump discharge pressure switch was found to be full of water.	This was caused by a ruptured tube most likely caused by wearout.	Replaced pressure switch and performed test.
RFW-LIC-620	Reactor Feedwater	With the plant at power, during routine observation of the Reactor Feedwater (RFW) startup valve Level Indicating Controller (LIC) meter went to zero when "tapped" by fingers. The performance of the system was unaffected.	The meter had an "open" in the circuit. Cause was unknown.	Replaced the Level Indicating Controller (LIC) and performed satisfactory retest.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-TR-614	Main Steam	While performing surveillance testing of the Main Steam (MS) Relief Valve Discharge Temperature Recorder (TR), with the plant at power, it was noted the CAM operated alarm switch actuated but the control room alarm did not annunciate.	Found a loose connection on the switch which actuates the annunciation.	Tightened loose connection and completed the surveillance test.
MS-RIS-610B	Radiation Monitoring	During a surveillance test, with the plant at power, Main Steam (MS) Line Radiation Indicating Switch (RIS) would not calibrate.	The radiation monitor drawer circuit was open. This was most likely due to age.	Replaced drawer with a spare and performed satisfactory test.
DLO-M-P/2A2	Diesel Lube Oil	During routine observation, with the plant operating at power, it was found that the motor operating the soak back pump for diesel generator engine #2 on the standby AC power system was running but the impeller was not turning.	The soak back pump had a broken shear pin on the motor shaft. Failure appeared to be normal wear.	Replaced pump motor coupling and shaft shear pin and returned to service.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DLO-M-P/6	Diesel Lube Oil	The motor driven oil pump (P/6) on the High Pressure Core Spray Diesel was observed to have high vibration.	The vibration was caused by a bearing failure caused by the previous failure of the pump to motor coupling.	Replaced motor with like kind.
RCIC-M-P/3	Reactor Core Isolation Cooling	The Reactor Core Isolation Cooling (RCIC) water leg pump was observed running noisy and vibrating.	The pump motor shaft was out of alignment.	Realigned Pump and replaced motor bearings.
RPS-M-MG1	Reactor Protection	A loss of power was experienced on Reactor Protection System (RPS) bus "A".	A motor bearing on the motor-generator set (RPS-MG-1) failed probably due to normal wear.	Replaced motor with spare and tested.
DLO-P-10	Diesel Lube Oil	During normal observation it was noted the #3 diesel generator lube oil soak back pump would trip on overload.	A motor brush was excessively worn causing reduced brush tension which resulted in brush to commutator arcing.	Replaced motor with spare and tested satisfactorily.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY RELATED EQUIPMENT (continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RCC-P-1B	Reactor Building Closed Cooling	An operator on tour noted the outboard bearing on Reactor Building Closed Cooling (RCC) Pump "1B" was running hot.	The cause was unknown but thought to be excessive lubrication or expected end of life for the bearing.	Bearing was replaced.
RFW-P-1B	Reactor Feedwater	During retest of the Reactor Feedwater Pump 1B (RFW-P-1B) following maintenance the thrust bearing overheated.	The inboard oil seal ring had been improperly installed and caused uneven loading on the bearing.	Replaced bearing and associated oil seal ring.
RFW-P-1B	Reactor Feedwater	With the plant at power a high vibration alarm was received on Reactor Feed Pump 1B (RFW-P-1B) in the control room.	A small orifice that admits oil to the thrust bearing was found plugged. The cause of the plugging was believed to be filter particles caused by filter changeout.	Bearing was replaced and lubrication drained. The procedure was changed to require draining prior to filter changeout.

2.5 INDICATIONS OF FAILED FUEL

INTRODUCTION

In accordance with the commitment and requirements described in the WNP-2 FSAR, Section 4.2.4.3, a visual inspection of discharged fuel from WNP-2, Cycle 4 was performed on October 5-10, 1989. The purpose of the inspection was to verify assembly and fuel rod structural integrity. In addition, although not a commitment, a visual inspection of selected discharged fuel channels was performed at the same time.

SUMMARY OF INSPECTION RESULTS

A total of ten assemblies and two channels discharged at the end of cycle 4 were inspected. No evidence of mechanical damage, geometric distortion or rod bow were observed. All rods inspected appeared properly seated in the lower tie plate. All spacers appeared to be in proper position. The fuel exhibited nodular corrosion which covered portions of the clad on fuel rods which were cleaned for inspection. The extent of coverage did not appear to be markedly changed from previous inspections although some instances of clad surface roughness were observed on profile. The assemblies, uncleaned, generally conformed to General Electric (G.E.) visual standard 2. However, in the single instance where a fuel rod was cleaned of surface crud, the observed nodular corrosion was substantially less than the 100% coverage associated with visual standard 2. Based on comparisons with end of cycle 3 fuel, it appears that nodular corrosion is still taking place but the rate of growth appears to be low.

Fretting marks appeared on several assemblies, particularly in the span 6 region. (See Results of Fuel Examination section for span location definition). Investigation as to the cause of the scratches in the 6th span has determined that they are caused by contact with the upper bracket of the WNP-2 south fuel preparation machine during de-channeling following discharge. The other scratches are assumed to be caused by foreign objects or by rubbing against the spent fuel storage locations during fuel movement. None of the scratches appeared to have sufficient depth to be of concern. Two of the inspected assemblies appear to have contained foreign material.

Some scratches were noticed on the bottom of the lower tie plate on some assemblies which might be indicative of a slight bundle rotation.

On some assemblies, the tie rods are apparently growing faster than adjacent fuel rods. This is causing an apparent loss of tension on the tie rod hex nuts.

One instance of finger spring damage was recorded on a photograph. The observed damage was most probably caused by fuel handling after de-channeling although spring relocation would have the same affect. The missions of the springs while in the core was not impacted.

Fuel rod E-4 of Fuel Assembly LTJ 511, which is the non-spacer/capture water rod, has what is either a clad imperfection or a foreign object wrapped around it at least for 270°. From the photographs, it is impossible to be differentiate further. The phenomenon occurs low on the rod in what is the natural enrichment zone of the core.

The inspected channels all exhibit a coating of flake-like oxide material. Some miscellaneous scratches were observed. There was no evidence of mechanical damage, holes in the channels or control rod shadowing effects.

SELECTION OF ASSEMBLIES AND CHANNELS

During the spring 1989 refueling outage, 136 original core fuel assemblies were discharged. Ten of these assemblies and two channels were selected for visual inspection. The ten assemblies represent greater than 5 percent of the discharged fuel and are representative of the highest burnup assemblies in the discharged batch. Visual examination of the peripheral fuel rods of these assemblies included observation for cladding defects, fretting, fuel rod bow, missing components, corrosion, deposition and geometric distortion. The selected assemblies are all high enriched (2.19 weight percent U-235; initial). The two channels selected were representative of the highest exposed channels discharged.

Some characteristics of the selected assemblies and channels are shown in Table 1.

TABLE 1.0

CYCLE 4 DISCHARGED FUEL ASSEMBLIES SELECTED FOR EXAMINATION

FUEL ASSEMBLY IDENTIFICATION	CHANNEL IDENTIFICATION	EXPOSURE (MWD/MT)	WET SIP TEST	ULTRA SONIC TEST	SUSPECT CELL
LJT 522	—	25,817	—	—	
LJT 770	—	25,217	X	X	X
LJT 398	—	26,026	X	—	
LJT 525	—	26,006	X	—	X
LJT 414	—	25,961	—	—	
LJT 713	—	26,209	—	—	
LJT 604	—	25,991	—	—	
LJT 511	—	25,971	X	—	X
LJT 737	71895*	26,083	—	—	
LJT 794	71473*	22,338	X	—	X

*The channel has the same exposure as the assembly it was on.

The ten assemblies inspected have exposures ranging from 22,338 to 26,209 MWD/MT. The inspected assemblies include assemblies which were sipped and, in some cases, ultrasonically tested for fuel leaks during the R-4 outage. In addition, some assemblies were located in fuel cells suspected of containing fuel leaks as determined from flux tilt testing.

INSPECTION TECHNIQUE

The poolside visual examination was performed with an underwater periscope system with results of the fuel inspection being recorded on the Nuclear Fuel Transfer List in addition to the inspectors working notebook. Two sides of each fuel assembly were viewed. Photographs of selected points of interest were taken. A total of eight-eight photographs of the examined fuel and channels were taken. Fifty-four of these photographs were successful. As the Nuclear Fuel Transfer List log and accompanying notes constitute the permanent record of the inspection, successful photographs of all inspected locations are not required although certainly desired. The inspection procedure involved moving the selected selected fuel assembly in a vertical direction past the fixed periscope. This was accomplished by raising the fuel assembly out of the spent fuel rack with the fuel handling mast on the refuel bridge. Channel inspection was performed in a similar manner. A piece of abrasive material was used to remove the heavy layer of red-colored surface crud from some of the edge fuel rods in order to assess the rate of nodular growth.

INSPECTION CRITERIA

Visual inspection of the selected fuel assemblies was performed according to the following criteria:

- Proper rod seating in the lower tie plate
- Rod bow and spacing
- Spacer location and perpendicularity
- Finger spring condition
- Condition of tie rod hex nuts and other structural components
- Nodular corrosion and crud scaling
- Fuel rod fretting

The channels were inspected for spallation, weld failures, cracks and other structural failures, and buildup of oxidation. The results are discussed below.

RESULTS OF THE FUEL EXAMINATION

With one possible exception, the inspected fuel assemblies exhibit good apparent integrity. The upper tie plates were level, fuel rod springs had ample compression space, the rod nuts appeared snug except in one or two instances and all the fuel rods observed were properly seated in the lower tie plate. The spacers appeared perpendicular to the fuel rods and were properly located. Most finger spring sets observed displayed no damage. Minor finger spring damage was observed in isolated cases. The grid spacers in general exhibit a heavy nodular buildup. Exceptions to the above statements along with specific phenomena observed on specific assemblies are discussed below on an assembly basis. The channels inspected displayed no instances of spallation, cracking or other loss of integrity. They did exhibit a heavy oxide corrosion covering on all non welded surfaces.

During the inspection, the Nuclear Fuel Transfer List was maintained, field notes were obtained and photographs were taken. Developed photographs were not obtained for all inspection points. During the inspection activities, it was discovered that the photograph taken of the first 5 assemblies were not correctly exposed. No attempt was made to re-examine and photograph these assemblies except for those cases where the inspection notes indicated a potential anomaly. The following description of the specific assembly inspection is based on the Nuclear Fuel Transfer List, the field notes and, where available, the photographs.

In discussing specific fuel assembly observations, the following convention will be used. With the threaded post of the assembly in the upper left corner, the top is side A, the right side B, the bottom side C and the left side D (See Figure 1). Fuel pin locations are identified as follows: With the threaded post in the upper left corner, the fuel rod columns are labeled from left to right A,B,C,D,E,F,G and H. The fuel rod rows are 1,2,3,4,5,6,7,8 from top to bottom. Fuel spacers are numbered 1 through 7 beginning at the lowest spacer and the regions between spacers, called spans, are numbered 1 through 8 beginning at the bottom of the fuel assembly.

Nodular corrosion was observed to some degree on all of the assemblies that were inspected. The assemblies, uncleaned, generally conformed to G.E. visual standard 2. However, in the single instance where a fuel rod was cleaned of surface crud, the observed nodular corrosion in no way approached the 100% nodular or sheet coverage associated with visual standard 2.

Fuel Assembly LJT 552 was inspected on sides A and B. No unusual mechanical features were observed. Scratch marks, later determined to be associated with de-channeling in the south preparation machine, were observed on both sides of the bundle just below grid 6 as has been seen before. The outer appearance of the fuel generally conforms to G.E. visual standard 2.

Fuel Assembly LJT 770 was inspected on sides A and D. No mechanical anomalies were observed. The uncleaned fuel conforms in appearance to G.E. visual standard 2. Preparation machine scratches were observed on both sides A & D just below grid 6. A small foreign object was observed at the span 4 height on fuel rod A-5 (side D).

Fuel Assembly LJT 398 was inspected on sides A and D. No mechanical anomalies were observed. The uncleaned fuel appears to conform in appearance to G.E. visual standard 2. Scratches made by the south fuel preparation machine were observed below grid 6 on side D.

Fuel Assembly LJT 525 was inspected on sides A and C. No unusual mechanical features were observed. The appearance of the uncleaned fuel assembly appears to be consistent with G.E. visual standard 2. Scratches were observed on side A, span 1 and span 5, which could have been caused by the spent fuel racks. Side A was the second side inspected after assembly de-channeling. Scratches were observed on the nose cone of the lower tie plate, side A, which appear to have been caused by a slight rotating movement of the assembly.

Fuel Assembly LJT 414 was inspected on sides C and D. No mechanical damage was observed. The appearance of the uncleaned fuel appears to be consistent with G.E. visual standard 2. A foreign object was observed on the finger springs on side D. Specific views were made of the lower tie plate, grid 5 and upper tie plate of side C and the lower tie plate, span 1 and span 4 of side D. The assembly was later re-inspected. On re-inspection, the foreign object was missing. Photographs were obtained of the lower tie plate span 5 and upper tie plate of side C and the lower tie plate, span 1 and span 4 of side D. Some finger spring damage was noted on side B as seen from side C. The type of damage observed is caused either by contact with a fuel rack after de-channeling or by a loss of spring tension. In either case, the mission of the finger springs while the fuel in the core was not impacted. Some evidence of minor bundle rotation can be seen on this photograph.

Fuel Assembly LJT 713 was inspected on sides A and D. No mechanical damage was observed. The appearance of the uncleaned fuel most closely matched G.E. visual standard 2. Preparation machine scratches were observed below grid 6 on span 6 on sides A and D. A hex nut is shown backed off on Side A. The backed off appearance is most probably caused by greater differential fuel rod growth of the tie rod. There does not appear to be a concern for loss of the nut. Photographs were taken of span 1, span 4, span 6 and the upper tie plate of side A and span 2 and span 6 of side D.

Fuel Assembly LJT 604 was inspected on sides A and D. No mechanical damage was observed. A hex nut on a tie rod is shown backed off on the view of side D. The appearance of the uncleaned fuel most closely matched G.E. visual standard 2. Preparation machine scratches were observed on span 6 of side A. Rotational type scratches can be observed on the lower tie plate view on side C. Photographs were taken of the lower tie plate and span 6 of side A and of the lower tie plate, span 3 and the upper tie plate of side D.

Fuel Assembly LJT 511 was inspected on sides A and D. The appearance of the uncleaned fuel most closely resembled G.E. visual standard 2. No mechanical damage was observed other than the specific phenomena discussed below. De-channeling scratches were observed in span 6 on both sides A and D. A roughness of the clad surface perhaps associated with enhanced corrosion may be observed in span 6 of side A. A foreign object or clad bulge was observed between fuel rod columns E and F as viewed from side A on rod E-4. This is the water rod which is not the spacer capture rod in the G.E. fuel design for the WNP-2 initial core. When viewed from side D, this same phenomenon can be

seen between fuel rod rows 3 and 4 and fuel rod rows 4 and 5. This object occurs in the span 1 region near the bottom of the lower end cap. This location appeared to be in the natural uranium blanked region of the core. Photographs of the lower tie plate, span 3 and span 6 of side A (two photographs of the lower tie plate region) and six photographs of span 1, span 6 and the upper tie plate (four photographs of span 1) were taken. This fuel assembly was located in a suspect cell during cycle 4 as determined by flux tilt testing.

Fuel Assembly LJT 737 was inspected on sides A and D. No mechanical damage was observed. The uncleaned fuel rods most closely conform to G.E. visual standard 2. Some clad roughness can be observed in profile on span 4 of side A. Preparation machine scratches can be observed on span 6 of side A. One of the hex nuts appears to have backed off a very small amount in the upper tie plate view of side D. Photographs were obtained for span 1, span 4 and span 6 of side A and span 1, span 5 and the upper tie plate of side D.

Fuel Assembly LJT 794 was inspected on sides A and D. No mechanical damage was observed. The fuel (uncleaned) most closely resembled G.E. visual standard 2. Some hex nuts loosening may be present. Photographs were taken of span 1, span 4 and the upper tie plate region of side A and the lower tie plate, span 4 and the upper tie plate of side D. Then a section of span 7 of side D was cleaned with an abrasive material (Scotch brite). After cleaning, some white oxide nodules could be observed on the clad surface. The nodular coverage is estimated at less than 30% which is less than the 100% coverage usually associated with visual standard 2.

Channel 71895, which has been resident on fuel assembly LJT 737 since initial startup, was inspected on sides A and D. This channel has an exposure of 26,083 MWD/MT and was measured for bow just prior to inspection. The channel passed the measurement criteria. No mechanical anomalies were observed on the channel. It was covered with a heavy uniform oxide layer, white in appearance, except for the weld seam, visible on side A, which exhibited occasional white oxide nodules. Photographs top of side A and the bottom and middle of side D were obtained.

Channel 71473, which has been resident on fuel assembly LJT 794 since initial startup, was inspected on sides A and D. This channel has an exposure of 22,338 MWD/MT and was measured for bow just prior to inspection. The channel passed the measurement criteria. No mechanical anomalies were observed on the channel. It was covered with a heavy white oxide layer except for the seam weld region. The seam weld visible on side A, had occasional white modules but was mostly clean of oxidation. Photographs of the top, middle and bottom regions of both side A and side D were obtained.

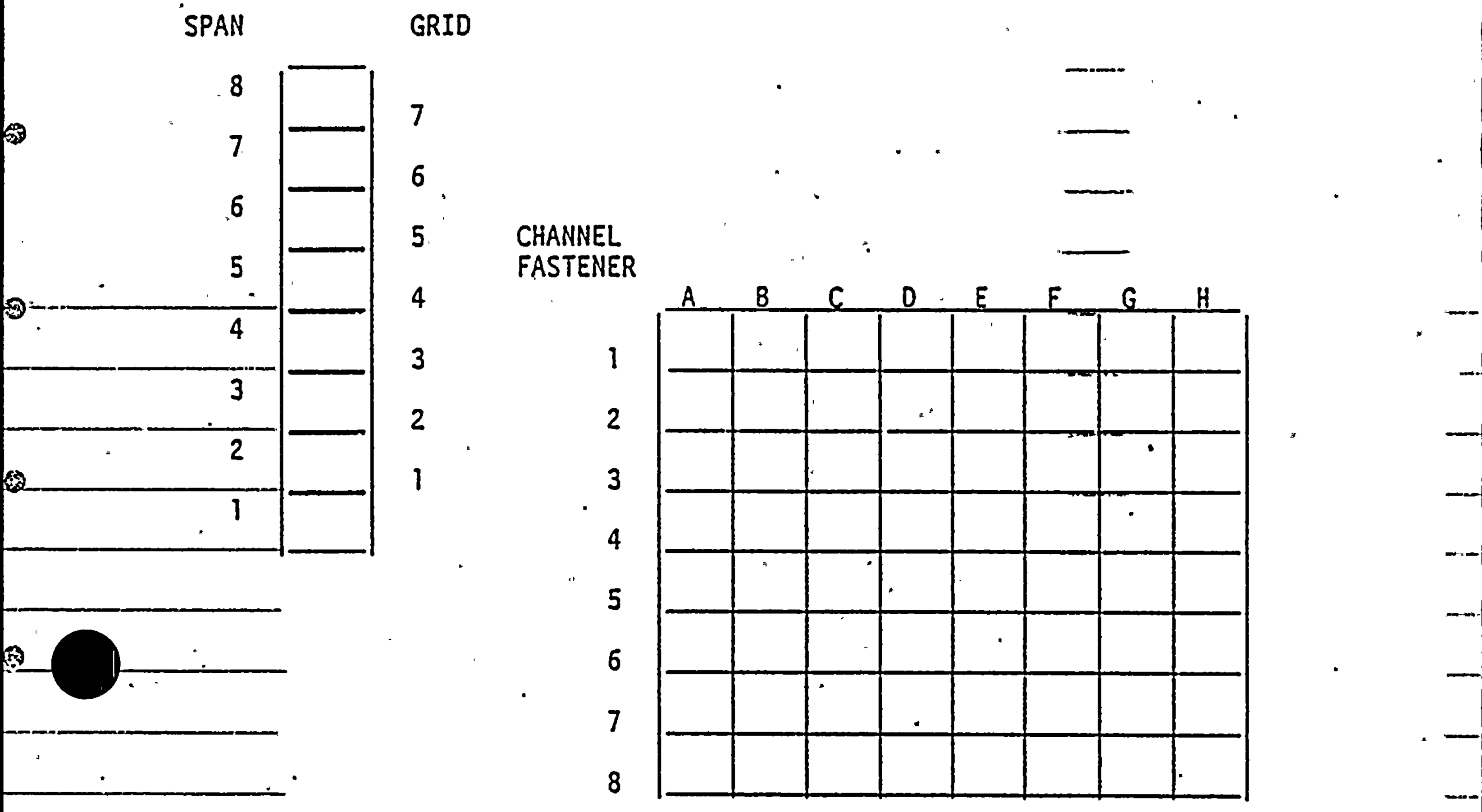


FIGURE 1. FUEL ASSEMBLY MAP SHOWING LABELING CONVENTION

2.6 PLANT MODIFICATIONS

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 1989 are provided. Included are summaries of the safety evaluations.

2.6.1 PLANT DESIGN CHANGES

The following plant design changes were completed in 1989 and reported in accordance with 10CFR50.59. These modifications were evaluated and it was determined that they did not (a) increase the probability of occurrence of an accident or malfunction of the equipment important to safety, as previously evaluated in the WNP-2 updated Final Safety Analysis Report (FSAR), (b) create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR, (c) reduce the margin of safety as defined in the basis for any WNP-2 Technical Specifications, or (d) require a change to the WNP-2 Technical Specifications and as such, prior NRC approval was not required.

Plant Design Change 84-0190

Plant Design Change 84-0190 was initiated to modify breaker control logic to allow operation of one Plant Service Water (TSW) pump during a LOCA when off-site power is available. This modification will minimize loss of TSW pumps and facilitate plant recovery from a LOCA.

To prevent an undesirable bus transfer due to voltage transients caused by large motor starting during a LOCA with power supplied from the startup transformer (TR-S), this design change provided ten second time delays to the automatic starts of ECCS pumps on a LOCA initiation. However, the start interval between ECCS pumps on the same division remained the same (i.e., 5 seconds). The safety analysis bounding times are unchanged. In addition, the automatic trip of SM-75 and SM-85 on a LOCA signal was defeated and automatic shedding of SM-72 and SM-82 on a LOCA signal was provided. Automatic trip of SM-75 and SM-85 on loss of offsite power was retained. As a result, this allows for continued operation of the TSW pumps during a LOCA and with offsite power available. Also, an electrical interlock was provided to prevent starting of the second TSW pump during a LOCA.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the margin of safety was not reduced or the possibility of a different malfunction as defined in the basis for any Technical Specification was not increased. Redundant safe shutdown equipment and systems will always remain operational and the required system response times were not affected. Qualitatively, the probability of a successful shutdown following a LOCA with offsite power available and a TSW pump available was increased, which qualitatively decreases the overall core damage risk.

Plant Design Change 84-0623

Plant Design Change 84-0623 was initiated to modify two Reactor Water Clean-up (RWCU) valves to decrease their stroke times for containment isolation. Previously, the valves were "blocked" from stroking full open to reduce the stem travel required to close. Blocking provided stroke times within the maximum allowable valve closure times.

This modification changed the design of the Limitorque operators to the RWCU-V-1 and RWCU-V-4 valves to increase the stroke speed of the valve. This design satisfied the maximum allowable stroke time without limiting the valve opening.

This modification did not result in a reduction in the margin of safety to the WNP-2 Technical Specifications or result in an unreviewed safety question because the valve closure times for the two RWCU containment isolation valves remained within the maximum allowable closure time.

Plant Design Change 84-1360

Plant Design Change 84-1360 was initiated to allow maintenance to be performed on one fire protection system in a given area without removing the fire protection alarm capability of a redundant system. This reduces the possibility of an undetected fire in areas where safety equipment is located during periods of maintenance on fire protection equipment.

This modification removed cross-connections between fire control panels FP-CP-FCP1 and FP-CP-FCP2. When an alarm is activated on one fire suppression system for a given area due to actual conditions or maintenance activities, the alarm covering the same area will remain functional to alarm on actual conditions only.

This modification did not involve a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the margin of safety in Technical Specifications was not reduced, and (2) this change provides for increased fire protection during maintenance activities.

Plant Design Change 85-0093

Plant Design Change 85-0093 was initiated to reduce the maintenance frequency on the Diesel Starting Air (DSA) system for the High Pressure Core Spray (HPCS) diesel engine and increase the reliability of the DSA to the HPCS diesel engine. Two high maintenance valves were removed from the DSA system which reduced the overall DSA maintenance. Also, DSA piping was rerouted to make redundant engine starting equipment completely independent. As a result, the overall reliability of the HPCS system was increased.

This modification removed the crosstie between air receivers DSA-AR-1C and DSA-AR-2C, which included a globe valve (DSA-V-5) and a check valve (DSA-V-6), respectively. A new 2-inch line was added to the line coming from air receiver DSA-AR-1C, another 2-inch crosstie line was removed, and a block valve (DSA-V-84) was added to an existing crosstie line to make redundant DSA equipment completely independent.

This modification did not result in a change to WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the modification increased the reliability of the DSA which increased to overall reliability of the HPCS system, and (2) the boundary conditions for the FSAR evaluations remained unchanged.

Plant Design Change 85-0184

Plant Design Change 85-0328 was initiated to increase the reliability of the portion of the leak detection system that monitors leakage from the reactor coolant pressure boundary. Monitoring is performed by sensing temperature increases and initiating alarms and isolations. The previous hardware had been causing an inordinate number of system isolations caused by spurious trips.

The old system (Riley Model 86) was replaced with a General Electric NUMAC system (LD-MON-1A, LD-MON-1B, LD-MON-2A, and LD-MON-2B). In addition, the system recorders (LD-TRS-608, LD-TRS-611, LD-TRS-622, and LD-TRS-624) were replaced with more reliable equipment. The new monitors provide automatic self-testing every 30 minutes that test all channels and functions of the monitor. In addition, there is constant monitoring for power failure and open T/C signal. The isolation logic and devices external to the temperature monitor units were not changed. A preoperational test was performed on the new equipment prior to return to service.

The change of hardware involving the leak detection system did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question evaluation concluded: (1) the function and performance of the Leak Detection did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

Plant Design Change 85-0328

Plant Design Change 85-0328 was initiated to remove a highly radioactive section of piping ("hot spot") in the drywell under the reactor pressure vessel. Removal of the "hot spot" significantly reduced radiation exposure to personnel performing maintenance activities in the immediate area, particularly on the control rod drives.

This modification removed a two-inch drain line (2" RRC(51)-1) between Reactor Recirculation (RRC) line 4"RRC(51)-4-3 and the Equipment Drains Radioactive (EDR) system header 4"EDR(47)-1 including valves RRC-V-29 and RRC-V-30. Caps were installed on the tees from the 4-inch RRC and EDR lines to maintain the reactor pressure boundary and seal off the opening to the EDR header, respectively. This line served no useful function during operation or shutdown. The line would ease draining of the reactor vessel during decommissioning but it is not required to achieve this draining.

Implementation of this modification was done through the use of a reactor vessel bottom head drain plug for isolation between the reactor vessel and the 2-inch RRC drain line. The bottom head drain plug must also perform as a pressure boundary for hydrostatic testing of the spool piece welds in the 4-inch RRC line to 1172 psig. The plug was back pressure tested to 1400 psig to demonstrate acceptability. A 10CFR50.59 evaluation determined there was no unresolved safety question related to the implementing activities or the plant configuration during implementation of this modification because: (1) the boundary conditions of the FSAR evaluations were not changed because a 2-inch leak through the bottom head drain at reactor shutdown conditions under atmospheric pressure are well within the postulated design basis conditions for the Small Break LOCA, and (2) the implementing activities did not reduce the margin of safety in the WNP-2 Technical Specifications.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) removal of the drain line and valves reduces the possibility of an inadvertent leak from the reactor pressure vessel, and (2) the boundary conditions of the FSAR evaluations were not changed.

Plant Design Change 85-0360

Plant Design Change 85-0360 was initiated to modify Class 1E and some non-1E 4.16 KV and 6.9 KV Westinghouse circuit breakers. A failed spot weld in the breaker linkage allowed the linkage to decouple. This had the effect of rendering electrical control circuits as well as anti-pump circuits (to prevent multiple breaker closures during faulted conditions) inoperable in affected breakers.

This design change fabricated and installed new linkage in all Class 1E 4.16 KV and 6.9 KV Reactor Recirculation Pump Westinghouse breakers. The new linkage piece used a pivot pin assembly that was plug welded instead of spot welded.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because this design change corrects a potential problem with auxiliary switch linkages for 4.16 KV and 6.9 KV Westinghouse breakers, and thereby, reduces the probability of a malfunction of equipment important to safety.

Plant Design Change 85-447 & 86-0557

Plant Design Changes 85-0447 and 86-0557 were initiated to increase the time delay in the ground fault relay settings to prevent spurious alarms from power transients.

This modification changed the time delay on the GRC type ground fault relays for motor control centers, and 4.16 KV and 6.9 KV switchgears from 2 cycles to 30 cycles. The ground fault relays provide alarm only and do not perform any safety function.

This modification did not result in a change to the WNP-2 Technical Specifications or result in an unreviewed safety question because: (1) the margin of safety was not reduced in the Technical Specifications, and (2) the boundary conditions of the FSAR evaluations remained unchanged.

Plant Design Change 86-0218

Plant Design Change 86-0218 was initiated to eliminate one of two fire suppression manual pull stations in the Communications Room (525-ft level) of the Radwaste Building because the station is inaccessible. The one remaining pull station in the area is much more accessible than the one removed.

This modification removed fire protection manual pull station FPMPS-28/31.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because this is not a safety-related system, the modification has no affect on safety systems, and removal of the pull station did not reduce the margin of safety in the Technical Specifications.

Plant Design Change 86-0332

Plant Design Change 86-0332 was initiated to provide increased assurance against overpressurization of the diesel fuel day tank for each of the three divisions due to failure of the transfer pump to stop on high level in the day tank. This increases the reliability of the diesel fuel oil system, thereby increasing the reliability of the diesel-generators.

This modification provided a passive overflow drain line between the diesel fuel oil day tank and its associated underground storage tank for each division.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall reliability of the diesel-generators was increased, and (2) the boundary conditions of the FSAR evaluations remained unchanged.

Plant Design Change 87-0031

Plant Design Change 87-0031 was initiated to modify motor-operated valve interlocks on the Residual Heat Removal (RHR) System to minimize the probability of inadvertent partial draining of the reactor pressure vessel to the suppression pool. The existing design did not pose a safety threat of completely draining the reactor pressure vessel because the water level would not drop below the top of the jet pumps.

This design change provided additional electrical control interlocks of the suppression pool spray and test return valves, RHR-V-24A & -24B and RHR-V-27A & -27B, respectively, with the RHR suction valves, RHR-V-6A & -6B. The existing interlock prevented opening of an RHR suction line valve if a suppression pool spray or test return valve in the same division is not fully closed. This modification provided interlocks against the reverse process. That is, the RHR suppression pool spray and test return valves in a given division are prevented from opening if the suction valve in the same division is open.

There were no modifications to the WNP-2 Technical Specifications as a result of this design change. This change did not involve an unreviewed safety question because the probability of maintaining a safe shutdown condition is increased and the margin of safety in the Technical Specifications was not reduced.

Plant Design Change 87-0114

Plant Design Change 87-0114 was initiated as a human factors improvement to minimize the possibility of operator error by changing the physical location of selected power bus control switches. Changing the switch locations made lineup of the switches relative to the sequence of manual operation consistent with all other similar power bus control switches, improving the human-to-control board interface. This modification reduces the possibility of a reactor scram due to operator error, which could occur if the switches are operated out of sequence.

This modification exchanged location of the following two pairs of switches on Board "C" that control power between the startup transformer TR-S and bus SH-6, and the normal transformer TR-N2 and SH-6: (1) synchronizing selector switches CB-S6 and CB-N2/6 exchanged locations, and (2) startup feeder CB-S6 and normal feeder CB-N2/6 exchanged locations.

This design change did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) all wiring remained the same; (2) only the location of switches changed, and (3) this modification reduces the possibility of a reactor scram due to operator error.

Plant Design Change 87-0316

Plant Design Change 87-0316 was initiated to provide annunciation to the main Control Room operators when the transfer switch for the second of two Residual Heat Removal (RHR) boundary isolation suction valves from the Reactor Pressure Vessel (RPV) is not in its required position of "Emergency". The new annunciator will alert operators that the RHR System is incorrectly lined-up and could lead to an overpressurization of the RHR System.

This modification changed existing wiring to energize an annunciator when the transfer switch for RHR-V-8 (switch number E-RMS-ARST24) is not in its required position of "Emergency" during Modes 1, 2, or 3. The transfer switch for RHR-V-8 must be in the "Emergency" position during normal operation to prevent it from inadvertently opening simultaneously with RHR-V-9 during certain postulated accident conditions. Simultaneous opening of the two valves during normal operation would lead to overpressurization of the RHR System.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall reliability of the RHR System was improved, and (2) the boundary conditions of the FSAR evaluations remained unchanged.

Plant Design Change 88-0038

Plant Design Change 88-0038 was initiated to replace selected high maintenance radiation and turbidity recorders with low maintenance recorders. Extensive manhours and spare parts were required to maintain the mechanical type recorders.

This modification replaced five existing recorders with Yokogawa recorders and installed one additional new Yokogawa recorder. The five replacement recorders consisted of one turbidity recorder on the Reactor Feedwater (RFW) system (RFW-TBR-622) and four radiation recorders on the Area Radiation Monitor (ARM) system (ARM-RR-600), Off-Gas (OG) system (OG-RR-601, OG-RR-604), and Reactor Building Exhaust Air (REA) system (REA-RR-603). The new recorder was installed for radiation recording on the Standby Service Water (SW) system (SW-RR-2).

These modifications did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the maintainability and reliability of the recorders were increased; (2) the boundary conditions of the FSAR were not changed, and (3) the margin of safety in the WNP-2 Technical Specifications was not reduced.

Plant Design Change 88-0056

Plant Design Change 88-0056 was initiated to add over pressure relief for the Reactor Building Outside Air (ROA) System. This prevents destructive overpressurization of the reactor building as had occurred on February 14, 1988.

This modification installed a relief damper for the ROA Heating and Ventilation Unit (HV) ROA-HV-1. In the event the Reactor Building Exhaust Air (REA) System fails to start or initiation lags that of the ROA System could result in increased reactor building pressure, the back draft damper provides a relief path back to the fan suction to prevent reactor building overpressurization. The relief damper is on the ROA fan intake and downstream of the ROA supply valves ROA-V-1 and ROA-V-2 that close under LOCA or radioactive release conditions. Thus, this modification does not compromise Secondary Containment.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because it did not affect a safety-related system, the modification to the ROA system did not change the boundary conditions used in the FSAR, and the WNP-2 Technical Specifications were not affected.

Plant Design Change 88-0306

Plant Design Change 88-0306 was initiated to provide increased assurance of appropriate Control Room HVAC System operation following a design basis Loss of Coolant Accident (LOCA), thus ensuring Control Room personnel post-event radiation doses remained within acceptable levels. During a LOCA, the normal fresh air intake for the Control Room HVAC is isolated and two remote air intake lines are opened. Each remote air intake line has two isolation valves with one valve powered from Division I and the other valve powered from Division II. In the unlikely event of a special single failure (i.e., "hot short" or "smart short") in a power division, a valve in each remote air intake line could isolate. With the loss of all fresh air input, the Control Room HVAC would continue to operate, but in the recirculation mode. In the recirculation mode, the Control Room would not remain pressurized with respect to surrounding areas. Acceptable post-event radiation doses to Control Room personnel could not be assured because operating post-LOCA in this mode was not analyzed. (This condition was discussed in LER 88-031.)

This modification replaced the motor operators on the four remote air intake isolation valves (WOA-V-51A, -51B, -52A, and -52B) with manual operators. This allowed one remote air intake line to be open continuously, thus assuring that a single failure could not cause operation in the recirculation mode. Also, post-event manual transfer could be made to the other remote air intake path if the currently open remote air intake path reached unacceptably high radiation levels.

This modification did not involve a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) changing the remote air intake valves to require manual operation eliminates the possibility of a single failure and ensures that the Control Room continues to meet the licensing design basis for analyzed radioactive dose rates; (2) no new event important to safety was created by this change, and (3) the margin of safety in the Technical Specifications was not reduced because one path will always be operational and the time required for Operator action has minimal dose impact.

Plant Design Change 88-0430

Plant Design Change 88-0430 was initiated to prevent premature failure of maintenance drain lines from two Main Steam (MS) trap stations. Drain valves were removed from each trap station and the lines capped to minimize flow induced vibration forces that were causing maintenance line fatigue failures. The drain lines were uncapped and the valves were replaced during the R-4 maintenance outage. Additional supports were provided to reduce vibration forces to acceptable levels.

This modification removed two Main Steam valves (MS-V-239 & MSV-238B) from Trap Station #2 drip leg piping and two Main Steam valves (MS-V-118C & MS-V-238C) from Trap Station #3 drip leg piping, and welded a cap to each of the respective drip legs. This resulted in temporarily disabling the drain capability of the trap stations until the maintenance outage. Draining of the trap stations can only be performed during shutdown and is normally done during the maintenance outage to remove built-up debris. As a result, this modification did not impact safety-related equipment nor increase the potential to degrade related equipment (e.g., main turbine).

Temporary removal of the valves did not require a modification to the WNP-2 Technical Specifications. This change did not involve an unreviewed safety question because the potential failure of the drip legs from vibration induced fatigue was reduced making the Main Steam system more reliable.

Plant Design Change 89-0141

Plant Design Change 89-0141 was initiated to ensure a Reactor Building pressure of -0.6 inch water gage within the existing HVAC system capability. The modification maintains adequate ventilation and cooling within all areas of the Reactor Building.

This modification changed the pitch of the blades of the Reactor Building Outside Air HVAC System fan ROA-FN-1A from a supply flowrate of 90,000 cfm to 70,000 cfm. With a building in-leakage of 5,000 cfm at -0.6 inch water gage and a nominal exhaust fan flowrate of 91,000 cfm for REA-FN-1B, the new supply fan configuration assures appropriate building pressure, even with moderate winds, without creating excessive loads on the exhaust fans. This extends equipment life and increases overall Plant reliability. Also, a building pressure of -0.25 inch water gage can be maintained under design basis conditions. Although the new air balance configuration reduces total ventilation flow below design, adequate HVAC is still provided for all areas of the Reactor Building. This is because the capability of HVAC system with the new air balance configuration exceeds the actual building heat load which was determined to be less than the design building heat load. In addition, the requirement to draw air from areas of minimum contamination through areas of higher contamination was satisfied.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) supporting calculations determined the HVAC system will meet the design basis requirements as described in the FSAR; (2) the boundary conditions of the FSAR evaluations were not changed, and (3) the margins of safety in the WNP-2 Technical Specifications were not reduced.

Plant Design Change 89-0178

Plant Design Change 89-0178 was initiated to reduce the time to energize the emergency buses (SM-7 & SM-8) from the backup emergency diesel-generators (DG). The relays that provide the contact permissive in the diesel-generator output breaker control circuit were electromechanical with marginal performance. More consistent DG start and load times can be realized with solid state relays.

This modification replaced the existing electromechanical GE relay DG-RLY-59 DG1/DG2 with an ASEA (ITE-27N) relay of solid state design to improve performance of the DG voltage permissive interlock for output breaker closure. This change will improve pickup voltage repeatability and provide faster and more consistent diesel start-to-load acceptance times.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the reliability and time to energize emergency buses from backup power was improved.

Plant Design Change 89-0200

Plant Design Change 89-0200 was initiated to minimize the possibility of containment liquid bypass leakage through the Control Rod Drive (CRD) System. Given failure of the CRD pumps, the existing design had the potential of releasing radionuclides in excess of the 10CFR 100 guidelines. This was based upon the design basis post Loss-of-Coolant Accident (LOCA) radiation dose calculations. This condition was identified as a result of a commitment made in LER 88-012 to evaluate WNP-2 for possible unmonitored release paths.

The design change installed two check valves (CRD-V-524 & -525), a globe valve (CRD-V-526), and three vent lines and valves upstream and between the two check valves and globe valve in the 2-inch CRD supply line upstream of two CRD filter units (CRD-FU3A & -3B). The check valves perform the safety-related function of preventing bypass leakage from the reactor vessel to the area outside of the reactor building during post-LOCA conditions.

This modification did not result in a change to the WNP-2 Technical Specifications or involve and unreviewed safety question because: (1) the probability of an unmonitored release from the CRD system was reduced; (2) the boundary conditions used in the FSAR evaluations were not affected, and (3) the margin of safety in the Technical Specifications was not reduced.

2.6.2 LIFTED LEADS AND JUMPERS

The following are summaries of noteworthy changes made in the facility by use of the Lifted Lead and Jumper (LLJ) Procedure (PPM 1.3.9) as required by 10CFR50.59. Each change was evaluated and determined not to represent an unreviewed safety question nor require a change to the WNP-2 technical specifications.

LLJ 289-207 (Change to SM-7 and SM-8 Minimum Bus Voltage Annunciation)

Problem Description

A review being performed in response to an Operational Experience Report (OER) discovered a problem with equipment powered from some distribution panels. This condition whould occur during plant conditions where bus voltage was slightly higher than the degraded voltage relay pick-up. An urgent Plant Modification Request was immediately processed to provide a permanent fix to this condition (See PMR 89-0159 under the Plant Modification Section of this report).

Discussion and Corrective Action

A Justification for Continued Operation was prepared which recommended that plant operators be made aware of the changes in the degraded voltage relay protection requirements. A Lifted Lead and Jumper-Temporary Modification was approved which changed the degraded voltage protection to provide annunciation in the control room upon the occurrence of the minimum acceptable voltage of 93%. In addition, plant annunciator procedures (PPMs 4.800.C1-2.4 and 4.800.C5-2.4) were modified to require operator action if the alarm occurred.

A 50.59 evaluation was performed to support this temporary change in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the undervoltage electrical protection met minimum requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-0221 (Temporary Power Provided to Division II 24VDC Battery Chargers)

Problem Description

During the Spring 1989 refueling outage transformers E-TR-8/81 and E-TR-8/83 needed to be taken out of service for Division II maintenance. With these transformers out of service the primary source of power to the Division II 24VDC system would be lost.

Discussion and Corrective Action

A Jumper and Lifted Lead request was processed and approved which allowed a temporary power supply from a non-Division II source to be connected to battery chargers E-CO-2A and 2B. This allowed continued operation of control room instrumentation as desired by plant operations and prevented the batteries from discharging during the Division II outage.

A 50.59 evaluation was performed to support this temporary change in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) maintaining the power to the Division II 24VDC system during the Division II outage did not change the function of the system (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-0222 (Temporary Power Provided to Division II 125VDC Battery Chargers)

Problem Description

During the Spring 1989 refueling outage transformers E-TR-8/81 and E-TR-8/83 needed to be taken out-of-service for maintenance. With these transformers out of service the primary source of power to the Division II 125VDC system would be lost.

Discussion and Corrective Action

A Jumper and Lifted Lead request was processed and approved which allowed a temporary power supply to be connected to distribution panel DP-S1-2 which allowed continued operation of control room instrumentation needed to monitor the safe shutdown status of the plant and prevent the B1-2 batteries from discharging. This was done by providing a jumper between the B1-7 and B1-2 batteries to allow the B1-7 charger to carry the load.

A 50.59 evaluation was performed to support this temporary change in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) maintaining the power to the Division II 125VDC system during the Division II outage did not change the function of the system (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-9999 (Temporary Power Provided to MC-7A)

Problem Description

During the Spring 1989 refueling outage transformer E-TR-7/73 needed to be taken out of service for Division I maintenance. With this transformer out of service the source of power to Motor Control Center MC-7A would be lost. This motor control center provides power to Division I plant monitoring instrumentation. Plant operating personnel requested power to allow monitoring activities to continue during the transformer outage.

Discussion and Corrective Action

A Jumper and Lifted Lead request was processed and approved which allowed a temporary power supply from a non-Division I source to be connected to MC-7A via SL-73. This maintained power to battery chargers C1-1 and C2-1 and allowed for continued operation of control room instrumentation as desired by plant operations.

A 50.59 evaluation was performed to support this temporary change in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) maintaining the power to the Division I MC-7A during the Division I outage did not change the function of the system (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-0224 (Change to Make "Bridge-Over-Core" Interlock Functional)
LLJ 289-0300

Problem Description

During the Spring 1989 Refueling Outage conductors 184 and 186 in the refueling bridge takeup reel were found broken on two different occasions. With these conductors broken the interlock for the "Bridge-Over-Core" was not functional.

Discussion and Corrective Action

Jumper and Lifted Lead requests were approved which allowed the use of power from a spare cable (SP-1) in place of the broken conductors. This provided power to the activity control unit logic to determine when the refuel bridge is in "Over-The-Core" status for implementation of refuel mode interlocks.

A 50.59 evaluation was performed to support these temporary changes in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the Refueling Bridge and its logic and interlocks did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ-289-0225 (Temporary Power Provided to Source and Intermediate Range Neutron Monitor Logic)

Problem Description

During the Spring 1989 Refueling Outage maintenance was required on the Division II safety-related Switchgear (SM-8). This, in turn, would cause a loss of power to the Source and Intermediate Range Neutron Monitoring (SRM and IRM) logic circuits. This was unacceptable since refuel mode surveillances were required which called for operation of the SRM/IRM logic.

Discussion and Corrective Action

A Jumper and Lifted Lead request was approved which allowed for temporary power to be supplied from a convenience outlet at the 522 foot level of the Reactor Building to the SRM/IRM logic. This allowed the control rod block to be cleared and the refuel mode surveillances to proceed during the outage.

A 50.59 evaluation was performed to support these temporary changes in the plant electrical configuration. The operation of the plant with this temporary power supply in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the SRM/IRM and its logic and interlocks did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-316 (One Main Steam Relief Valve Declared Inoperable Due to Inadequate Air Supply)

Problem Description

During the Spring 1989 refueling outage the flex-hose air supply (CIA-FLX-1C) to Main Steam Relief Valve (MS-RV-2D) was found damaged beyond repair. The flex-hose could not be replaced prior to plant startup.

Discussion and Corrective Action

A Lifted Lead and Jumper request was approved which removed the flex-hose from the relief valve and replaced it with a blind flange. Thus, the manual relief (air actuated) function of the valve was not operational. The safety (spring lift) function of the valve is still operational.

A 50.59 evaluation was performed to support this temporary change in the plant mechanical configuration. The operation of the plant with this temporary flange in place did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation and function of the relief and safety valves for the primary pressure boundary did not change since only twelve of the eighteen valves are required to be operational (In addition, this valve is not one of the Automatic Depressurization System Valves), (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-469 (Defeat of Alarm "Remote Shutdown or Alternate Remote Shutdown Transfer Switch Activated")

Problem Description

During the Spring 1989 refueling outage two fans in the Radwaste Building (WMA-FN-52B and WMA-FN-53B) were being operated from Fire Remote Transfer Panel 1 (FRTP1) because of degraded voltage concerns. When the fans' control switch is placed in EMERGENCY to operate the fans, an alarm is generated to signal the control switch is not in NORMAL. This masks all other alarms that could be generated if other NORMAL/EMERGENCY control switches were placed in EMERGENCY.

Discussion and Corrective Action

The alarm was occurring because Fire Remote Transfer Switch (E-RMS-FRTS-5) was in emergency to allow operation of the two fans from FRTP1. A jumper and lifted lead request was approved which deactivated the alarm from E-RMS-FRTS-5.

The defeat of the alarm from E-RMS-FRTS-5 did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) restoring the annunciator to a usable state met all requirements and allowed monitoring of the remaining remote and alternate remote shutdown panel switches, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

LLJ 289-0493 (Temporary Jumper to Allow Standby Service Water Loop "A" to Remain Operational With the Pump Discharge Valve (SW-V-2A) Non-operational in the Full Open Position)

Problem Description

While starting the Standby Service Water System "A" the pump discharge valve (SW-V-2A) failed to open. Further investigation found the valve operator motor still running but no longer engaged to the operator. The valve was manually placed in the full open position and Standby Service Water Loop "A" continued to operate. However, in this condition, if the service water pump (SW-P-1A) were to trip (e.g., loss of off-site power) the pump would not be able to start since it needs a "SW-V-2A CLOSED" permissive to start.

Discussion and Corrective Action

A Jumper/Lifted Lead request was approved which defeated the "SWV-2A CLOSED" permissive. The use of the Standby Service Water Systems without the "SW-V-2A CLOSED" permissive did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question concluded: (1) the performance of the Service Water System met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

2.6.3 FSAR AMENDMENT EVALUATIONS

The following are summaries of changes made to the FSAR in Amendment 40 which were not initiated as a result of a plant modification. As part of the process of submitting an FSAR change, an analysis is performed in accordance with 10CFR50.59 to ensure the proposed modification does not involve an unreviewed safety question. The following summaries represent changes in system operation, clarification and/or updates of system descriptions, clarification of Supply System positions and, in some cases, changes to commitments previously made in the FSAR.

Chapter 9, Standby Service Water

MODIFICATION - This revision to the FSAR changes the requirement for minimum cooling water flow to the Residual Heat Removal (RHR) Loop C pump (RHR-P-2C) seal from 9 gpm to 0 gpm.

Basis For Change - This change was based in part on the similarity in design and operating conditions during a design basis accident between the RHR Loop C pump and the Low Pressure Core Spray (LPCS) pumps. The design and size of the seals are very similar between the LPCS and RHR pumps (i.e., 3.5 inch OD shaft versus 3.75 inch OD shaft, respectively). The LPCS pump and RHR Loop C pump seal flushing operating temperature conditions are the same (i.e., maximum normal operating water temperature is 120°F with a peak temperature of 212°F for accident temperature). During the Reactor Pressure Vessel (RPV) cooling mode, the RHR Loops A and B pump seal flushing line suction water temperature reaches 335°F. Since the LPCS specification does not require cooling water for its seal and the RHR Loop C pump seal does not experience the high temperature fluid that Loops A and B do, the Loop C RHR pump does not require any cooling water flow.

2.6.4 OTHER

The Plant Problems-Plant Problem Reports Procedure (PPM 1.3.15) provides instructions for the disposition and documentation of plant problems. An immediate disposition using the "Use-As-Is" or "Repair" options is considered a "change" within the definition of 10CFR50.59. Each item below has been evaluated to provide assurance that the disposition does not involve a change to the technical specifications or an unreviewed safety question.

NCR 288-0356 (Maintenance of Secondary Containment Negative Pressure)
NCR 288-0357

Problem Description

The FSAR (6.5.1 and (9.4.2) and the Technical Specifications (3/4.6.5) require the secondary containment (reactor building air space) to be less than .25 inches of vacuum water gauge. The pressure devices which measure this limit did not compensate for the environmental effects of differential temperature and could have resulted in a situation where the vacuum limit would not be maintained for secondary containment. In addition, there is a documented concern regarding the re-establishment of secondary containment differential pressure following a design basis accident.

Discussion and Corrective Action

A Justification for Continued Operation (JCO) was prepared which concluded the existing setpoint for the pressure measuring devices of .60 inches of vacuum water gauge would accommodate the environmental effects and maintain the required vacuum in the secondary containment during normal operation. A second JCO was prepared to justify operation of the plant while calculations are completed on both offsite and onsite doses during postulated design basis accident conditions with new secondary containment assumptions.

As a consequence of the evaluations performed in preparing the second JCO, the Standby Gas Treatment System (SGTS) flow was increased and credit was taken for building inleakage less than the Technical Specification limits resulting in an unreviewed safety question. The unreviewed safety question evaluation concluded the function of the secondary containment could be maintained and all study calculations show the offsite and onsite doses to be below 10CFR100 limits following design basis accidents. Ultimate resolution of this problem will involve Technical Specification and FSAR changes and require significant calculational updates.

PER 289-0009 (Emergency Lighting Failure During Annual Discharge Test)

Problem Description

Several eight hour emergency battery lights failed their annual discharge test being conducted by plant surveillance procedure (PPM 10.25.63).

Discussion and Corrective Action

Battery units were replaced to the extent permitted by available spares. A JCO was prepared which concluded that sufficient emergency lighting was available for operation, access and egress. This included an evaluation of physical lighting installed and functional, and a drawing review to ensure lighting was provided in all necessary areas. The disposition of this item was "Use-As-Is".

The use of the Emergency Lighting System as-is did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question evaluation concluded: (1) the function and performance of the Lighting System did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-019 (Identification of Four New Failure Modes for the Containment Nitrogen System)

Problem Description

Four new failure modes for the Containment Nitrogen (CN) System were identified that should have been analyzed as part of the plant design including: (1) A postulated break in the Auxiliary Steam piping, (2) Swamping the low flow vaporizer, (3) Design Basis Tornado, and (4) Rupture of the Nitrogen Storage Tank or its associated piping.

Discussion and Corrective Action

Each of the identified failure modes were analyzed and a Justification for Continued Operation was completed. The immediate disposition of this item was "Use-As-Is" with a deviation to two plant procedures and the placement of a portable alarming oxygen monitor in the control room under certain conditions.

The use of the Containment Nitrogen System as designed and constructed did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question evaluation concluded: (1) the potential for damage to plant equipment and the containment was very low, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-020 (Secondary Fuse Covers Installed in Safety-Related Motor Control Centers Without Proper Design Control)

Problem Description

Personnel safety secondary fuse covers were installed in selected safety-related motor control centers without proper design control. The covers were installed to alleviate a personnel shock hazard in some 480 volt motor control centers. No Plant Modification was processed and no 10CFR50.59 evaluation was performed to evaluate the change.

Discussion and Corrective Action

A Maintenance Work Request (MWR) was initiated to inspect and record the type of covers used in each location. A 10CFR50.59 evaluation was performed which concluded: (1) that the probability of occurrence or the consequences of an accident or malfunction of equipment as evaluated in the FSAR would not be increased because the fuse covers were fabricated of insulation material which cannot present any electrical failure mode and their light construction prevented the possibility of seismic concerns, (2) there was no possibility of creating an accident or malfunction of a different type than evaluated previously in the FSAR because the covers provided added insulation in the area of the fuse blocks which add to the safety of the design by reducing the possibility of failure during accident conditions, and (3) the margin of safety as defined in the Technical Specifications was not reduced.

In addition, the process for performing plant modification was changed to clearly require a formal modification before physical changes are initiated in the plant.

PER 289-026 (Gasket of Incorrect Thickness Installed in Several Main Steam Safety/Relief Valves)

Problem Description

The vendor for the Main Steam Safety/Relief Valves specifies a 0.125 inch thick eductor (bonnet) gasket. The Supply System's Materials Management System incorrectly specified a 0.250 inch thickness for the gasket and these were procured and installed in the plant. When this problem was discovered in January 1989 eight incorrect gaskets were still in stock in the warehouse. Records showed that at least 10 of the incorrect gaskets had been withdrawn from spare stock previously, of which four were withdrawn from the warehouse and installed on spare valves, and six were withdrawn from the warehouse and installed on in-service valves in the plant.

Discussion and Corrective Action

The immediate disposition for this item for the installed in-service valves was to "Use-As-Is" based on the following actions: (1) the vendor (Crosby) was contacted to identify potential concerns associated with the incorrect gasket thickness, (2) it was concluded that the impact on the setpoint would be in the conservative direction, (3) blowout of the gasket would not be likely because of a groove machined in the body, (4) any increased leakage would not be a problem as it would be identified and dispositioned in accordance with existing procedures, (5) misalignment of the body-to-bonnet joint would not be a problem since the alignment at the joint is controlled by the diametral fit of the eductor in the body, and (6) valve function using the air actuator would not be affected by the thicker gasket. The incorrect installed gaskets will be replaced per the normal preventative maintenance schedule.

Other corrective actions were taken as follows: (1) the incorrect "Material Code" has been deleted and replaced by the correct code, (2) new gaskets were ordered and the incorrect gaskets were scrapped, and (3) the correct gaskets were installed in the spare valves.

The use of a gasket of incorrect thickness did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the valve function and performance did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-029 (Limit Switches and Connectors on Containment (Wetwell-Drywell) Vacuum Breaker Valves Installed Without Proper Seismic and Environmental Qualification Review of the Design Change)

Problem Description

Limit switches and connectors for nine containment (wetwell-drywell) vacuum breakers were installed without proper seismic and environmental qualification review of the design change. The plant modification changed the type and mounting of the position switches and added a CONAX connector for the wires exiting the valve between the two discs. The connectors constitute part of the wetwell/drywell isolation as they penetrate between the dual disks of each wetwell/drywell vacuum breaker.

Discussion and Corrective Action

The immediate disposition for this item was "Use-As-Is". The design change was reissued and reviewed to quality class I requirements. The review showed the new switches were mounted with two more bolts than the original switch and the new switch had less mass than the old. Since the switch itself has no safety-related function (they are for indication only) the evaluation was limited to a seismic review of the mounting which met all requirements. The connectors themselves were found to be specified to quality class I requirements. The seal uses ceramic separators with Grayfoil packing to prevent leakage and leakage tests were performed on the seals.

The use of the limit switches and connectors did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the wetwell-drywell vacuum breaker function and performance did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 89-033 (Residual Heat Removal Heat Exchanger Thermal Relief Valve Installed Backwards)

Problem Description

The Residual Heat Removal Heat Exchanger (RHR-HX-1A) service water (tube side) thermal relief valve (SW-RV-1A) was found to be installed backwards. The relief valve inlet was bolted to the discharge piping and the relief valve discharge was bolted to the heat exchanger tap. SW-RV-1A provides thermal overpressure protection to RHR-HX-1A if the tube side is isolated by its block valves (RHR-V-14A and RHR-V-68A). This heat exchanger is used for shutdown cooling and also functions as part of the Emergency Core Cooling System (ECCS).

Discussion and Corrective Action

The immediate disposition for this item was "Use-As-Is" until the system was available to reposition the valve during the next outage. Until that time thermal relief protection was provided by tagging open the service water isolation valve (RHR-V-14A). With this valve tagged open thermal overpressure of the heat exchanger could not occur.

The use of SW-RV-1A in the backwards configuration did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the RHR-HX-1A function and performance did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-038 (Average Power Range Monitor Channel F Placed in Bypass)

Problem Description

In January 21, 1989 the Plant received a half scram caused by a loose K18 relay socket. The relay socket had been loosened by repeated removal of the relay in accordance with Average Power Range Monitor (APRM) surveillance procedures.

Discussion and Corrective Action

A 10CFR50.59 Safety Evaluation was performed to allow APRM Channel F to be bypassed until the next outage when the defective relay socket was replaced. The APRM bypass selector switch for channels B, D, and F was caution tagged to select APRM F for bypass except while performing surveillance tests on APRM B or APRM D. Operation with a single APRM bypassed was consistent with the FSAR and the Technical Specifications. A plant shutdown occurred on January 30, 1989 and the K18 relay socket was replaced on January 31, 1989.

The use of the APRM System with Channel F placed in Bypass did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the APRM function and performance did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0041 (Check valves found installed backwards in Diesel Starting System)

Problem Description

In January 1989 seven check valves were found installed backwards in the Diesel Starting Air Systems for DG1 and DG2. The valves and attached lines were painted in line and appeared to have been installed incorrectly in the factory. The valves are spring-ball check valves and are located in the bypass line that connects the air supply line to the starter pinions and to the line leading between the pinions and the air start relay valve. The incorrect valve direction was noted by Quality Assurance on a routine walkdown of the system.

Discussion and Corrective Action

A Justification For Continued Operations (JCO) was prepared and the immediate disposition was "Use-As-Is". The JCO showed that the check valves in the air start logic provided no essential function and did not impact the operability of the diesel generator units. This was due to the vent path of the external air port of the upper air start motor pinion. In addition, there was no indication of any malfunction in the air start system during several hundred starts performed in the factory and during plant startup and operation. The check valves were installed in the correct orientation during the next refueling outage in May 1989.

The use of the Diesel Starting Air check valves in the backwards orientation did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the Diesel Generator function and performance did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0094 (Failure of a Damper Motor in the Diesel Generator Heating and Ventilating System)

Problem Description

A damper motor in the Diesel Mixed Air System (DMA-AD-51) failed in the recirculation position. This damper is an outdoor mixing damper for the air handling unit which cools Division II cable and equipment in the corridor during diesel operation.

Discussion and Corrective Action

This item was dispositioned "Use-As-Is". A JCO was prepared which included a calculation by Engineering which showed that adequate cooling was provided with the damper in the failed position.

The use of the Diesel Mixed Air System with the failed damper did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the system met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0098 (Standby Service Water Pumphouse Ash Filters Not Installed in Accordance With Design Requirements)

Problem Description

A review of the Standby Service Water Pumphouse Heating and Ventilating System revealed three deficiencies with regard to the ash fall filters needed to protect against the design basis volcanic event. WNP-2 has two standby service water pumphouses (an "A" and a "B" pumphouse) and each pumphouse has two sets of filters. For one set of filters in each pumphouse (PRA-FL-2A/2B) the ashfall filter boxes were not accessible as they were blanked off by installed sheet metal. The second set of filters in each pumphouse (PRA-FL-1A/1B) were actually installed in the filter boxes contrary to design requirements which call for filter installation only under abnormal conditions (ashfall). The third deficiency was the requirement in the plant procedures for replacement of the filters every three hours or when the delta P indication across the filters exceeds a predetermined value. The delta P indicators were never installed.

Discussion and Corrective Action

The immediate disposition for this item was "Use-As-Is". For PRAFL-2A/2B the sheet metal was removed and the filter boxes are now accessible. For PRA-FL-1A/1B the filters were removed and placed in standby status. The requirement for delta P indication was removed from the plant procedure. The plant operators are required to replace the filters every three hours in the event of an ashfall. Calculations show that the three hour changeout time is very conservative.

The use of the Standby Service Water Pumphouse ashfall filters in the "as-is" configuration did not result in a change to the WNP2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Service Water System was not degraded, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0179 (Calculated Non-Conservative Doses to Control Room Operators
NCR 288-0403 Post-LOCA)

Problem Description

Under post-LOCA conditions engineering calculation NE-02-88-27 (performed in support of NCR 288-0403 took credit for 100% mixing of primary containment leakage within the reactor building volume before postulating a release to the environment through the standby gas treatment system. This assumption was in conflict with Regulatory Guide 1.3 and resulted in a non-conservative dose estimate for the control room under accident conditions.

Discussion and Corrective Action

After an evaluation by Plant Management the Shift Manager declared an Unusual Event and a controlled shutdown of the plant commenced. After three hours the shutdown was halted at 52 percent power consistent with the Engineering analysis indicating that the associated reduction in source term was adequate to ensure habitability while the calculation problem was being resolved. Compensatory measures were defined to assure control room habitability, including the requirement that both control room remote air intakes remain open to assure control room habitability, an operator be dedicated to respond within 20 minutes to close one of the remote air intakes in the case of high radiation; and that system operating procedures be modified to reflect the new restrictions. This problem was resolved by improving the calculational methodology and removing unnecessary conservatism.

A 50.59 evaluation was performed to support the continued operation of the plant at full power. The operation of the plant with the revised analysis and the compensatory measures for control room ventilation operation did not require a change to the WNP-2 Technical Specifications or involve an unreviewed safety question. Resolution of the associated NCR (288-0403) on single failure vulnerability of the control room remote intakes did necessitate a Technical Specification change to exit the action statement requiring the control room pressurization mode of operation.

PER 289-0487 (Temporary Removal of Service Water Valves Associated with
LLJ 289-0353 Diesel Cooling Water)
LLJ 289-0376

Problem Description

Surveillance testing on the Division I Diesel Generator showed increasing high temperature in the diesel cooling water. This event was traced to the failure of service water inlet isolation valve SW-V-214 to properly open. This valve is in the line that supplies water to one of two Diesel Cooling Water (DCW) heat exchangers.

Discussion and Corrective Action

A root cause analysis of the failure of SW-V-214 determined that the disc to shaft taper pins had corroded and subsequently worked loose. The recommended action was to remove this valve and the other three valves of the same design and application (SW-V-215, 216, and 217) to preclude the potential for similar failures in the future. Jumper and lifted lead requests were approved which replaced each of the four valves with straight "spools". Other service water valves (SW-V-4A and SW-V-4B) will be used for heat exchanger isolation.

The use of the Diesel Cooling Water and Standby Service Water Systems without the four isolation valves did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Division I and II Diesels were unaffected, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0573 (Temporary Change to Allow Repairs On the High Pressure Core
LLJ 289-0429 Spray (HPCS) Air Compressor Diesel (DSA-ENG-C/2C))

PDF 289-0590

Problem Description

The air start diesel (DSA-ENG-C/2C) on the HPCS diesel Starting Air System (DSA) was not functioning correctly as it would not shut down after an auto start.

Discussion and Corrective Action

A jumper and lifted lead request was approved which disabled the compressor and plant procedures were deviated to allow for operation of the air start diesel in an emergency.

The use of the Diesel Air Start System with the Diesel Compressor disabled did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the HPCS met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0588 (Inadequate Service Water Flow Through Critical Switchgear Air Handling Unit)

Problem Description

Service water flow through critical switchgear air handling unit cooling coil WMA-CC-53B1 could not be adjusted to a value greater than 58 GPM. FSAR Table 9.2-5 requires a minimum flow of 60 GPM. Partial blockage of the piping and/or cooling coils is indicated.

Discussion and Corrective Action

A justification for continued operation was prepared and a review of the Engineering calculation showed a minimum flow of 54 GPM would provide adequate cooling.

The operation of WMA-CC-53B1 with slightly reduced flow did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the cooling of the critical switchgear rooms met minimum requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0649 (Recirculation Flow Control Valve Penetration Transmitted Vibration and Noise)

Problem Description

At the 83% open position recirulation flow control valve (RRC- FCV-60B) penetrations transmitted vibrations and noise. The noise was noted in the northwest corner of the 501 foot elevation of the reactor building and the hydraulic lines to Recirculation Cooling Pump "B" were vibrating and noisy.

Discussion and Corrective Action

The valve was opened to the full open position and the vibration and noise stopped. Flow and power traces obtained from the Transient Data Acquisition System were reviewed by Engineering. Copies of the data traces were submitted to General Electric for review. At the next outage entry was made into containment and the valve and the area around the valve was inspected. All equipment appeared to be undamaged and operated normally.

Plant operation with the noise and vibration did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Recirculation System met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0650 (Change to Reactor Recirculation Flow Control Valves Runback Limit Setpoint)
PDF 289-0653
ISCR-937

Problem Description

A reactor scram occurred from 100% power when one of the Reactor Feed Pumps (RFW-P-1B) tripped. The scram occurred on low water level since the remaining feed pump was not able to maintain vessel level. The problem was traced to an inappropriate Reactor Recirculation (RRC) runback setpoint.

Discussion and Corrective Action

The procedure was deviated and the setpoint for the RRC flow control valves (RRC-FCV-60A/B) was changed from the incorrect 30% open position to the correct 20% open position. This setpoint had been verified during plant startup as the correct value to allow for recovery from a feedpump trip.

Plant operation with the revised flow control valve setpoint did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Recirculation and Level Control Systems met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0736 (Incorrect Duty Cycles for Safety-Related 125VDC Batteries)

Problem Description

The Supply System's internal Safety System Functional Inspection (SSFI) discovered an incorrect assumption in the calculation of duty cycles for the Division I and II Safety-Related 125VDC Batteries. When calculations in breaker actuation sequencing were made it was incorrectly assumed that the spring charging motors associated with the 480VAC switchgear were energized after "closing" as is the case with 4160VAC switchgear. However, the 480VAC switchgear motors are energized after breaker "Trip".

Discussion and Corrective Action

A Justification for Continues Operation was prepared and approved. The 480VAC breaker closing spring charging motors added a 10 second load during the first minute of battery discharge of 50 Amps for Battery B1-1 and 60 Amps for Battery B1-2. The capacity requirement for these batteries is determined by the steady state loads (two-hour) not by the first minute loading. Therefore, adding the 480VAC breaker spring charging motors to the first minute load did not change the battery capacity requirement.

Plant operation with the existing Division I and II 125VDC Batteries did result in a change to the WNP-2 Technical Specifications and resulted in an unreviewed safety question evaluation which showed: (1) the existing batteries are capable of supplying the updated battery duty cycles, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0747 (Inadequate Electrical Separation and Non-Failsafe Design of the Reactor Building Exhaust Air Radiation Monitoring System)

Problem Description

During the preparation of a Plant Modification three discrepancies were discovered in the Reactor Building Exhaust Air (REA) radiation monitoring system. They consisted of inadequate physical separation in Control Room cabinets, routing of failsafe cable in non-failsafe raceways outside of the control room, and a non-failsafe design response of the radiation monitors to inoperative/downscale conditions.

Discussion and Corrective Action

A justification for continued operation was prepared and approved by the Plant Manager. The failsafe circuits routed in non-failsafe raceways were placed on an hourly fire tour to minimize the probability of a fire that could cause a circuit fault and the REA radiation monitor downscale annunciator response procedure was revised to require operator action to place the affected trip monitor in a tripped condition upon receipt of a valid downscale condition. An engineering evaluation and a plant modificaion are being prepared to provide a permanent change to correct the problem.

Plant operation with the REA radiation monitoring system "as-is" did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Radiation Monitoring System met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0860 (Plant Operation With RHR-V-40 Tagged in the Closed Position)
PDF 289-0868

Problem Description

Generic letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance", caused a review of motor operated valve operability. The results of this review showed that one of the Residual Heat Removal (RHR) Loop "B" Discharge Valves from the Supression Pool (RHR-V-40) to the main turbine condenser had a motor operator that did not provide sufficient starting torque at degraded voltage to operate at the maximum design differential pressures.

Discussion and Corrective Action

A Justification for Continued Operation was prepared and approved which placed RHR-V-40 in a closed danger tagged position. Manual handwheel closure of the valve is performed after each opening. In addition, RHR-RLY-80/V40, the relay that activates the Bypassed and Inoperable Status Indication (BISI) for a series of motor operated valves was removed to clear the BISI alarm associated with this valve.

Plant operation with RHR-V-40 in a closed danger tagged position did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question evaluation concluded: (1) the overall operation of the Residual Heat Removal System did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PER 289-0869 (Diesel Generator Room Overtemperature Conditions During Postulated Accident Conditions)

Problem Description

Calculation of diesel generator room ambient temperatures exceeded values stated in the FSAR based on new diesel heat loads determined from a 24-hour test. Limiting temperatures were based on postulated accident conditions involving a Loss of Offsite Power during ashfall conditions.

Discussion and Corrective Action

A justification for continued operation (JCO) was performed which showed that the plant could operate until the Spring 1990 refueling outage (through April) without changes to the diesel room cooling system. The JCO was based on the fact that the hot weather that provides the limiting condition for the temperatures will not occur during that time period.

Plant operation with the Diesel Room cooling "as-is" did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the diesel room cooling systems will meet all requirements during winter and spring conditions, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed during the period of operation.

2.7 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.

PARTIAL DRAINING OF THE SPENT FUEL STORAGE POOL (TEMPORARY PROCEDURE 2.8.14)

A temporary procedure was written to lower the spent fuel pool level 19 inches to allow check valve maintenance to be performed. The fuel pool diffuser check valves FPC-V-146A and FPC-V-146B required modifications to their internals to implement an Equipment Modification Specification.

The controlled lowering of the spent fuel pool level by 19 inches did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the fuel pool cooling systems met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed during the period of operation.

WIDE RANGE NEUTRON MONITOR FINAL TESTS (PPM 8.3.123 AND 8.3.74)

Final testing was completed on the Wide Range Neutron Monitor System installed per the requirements of Licensing Condition 16 and Regulatory Guide 1.97. The pressure integrity of the in-containment cable assemby was verified using test procedure PPM 8.3.123. During the subsequent startup the system was calibrated and an operability check was performed in accordance with plant procedure PPM 8.3.74.

The performance of this final test did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Wide Range Neutron Monitoring System met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

ROD WORTH MINIMIZER (RWM) PREOPERATIONAL TEST (PPM 8.6.11)

A preoperational test was performed on the Rod Worth Minimizer (RWM) following the modifications made by Computer Change Request (CCR) 89-001. The test verified the operation of the replacement Rod Position Information System (RPIS) interface boards and the software change made to the existing Input/Output subroutines. The modifications resolved the problem on the RWM with the Select Error Indication in the Transition Zone (greater than Low Power Set Point (LPSP) and less than Low Power Alarm Point (LPAP)).

The performance of this preoperational test did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the performance of the Rod Worth Minimizer met all requirements, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

2.8 PLANT PROCEDURE CHANGES

The Plant Procedure control program requires a 10CFR50.59 evaluation whenever a procedure is changed which provides assurance that the disposition does not involve a change to the technical specifications or an unreviewed safety question. The following are summaries of significant Plant Procedure changes processed during 1989:

PDF 289-0152 (Procedure Change to Allow the Reactor Water Cleanup System (RWCU) to Operate During Modes 4 and 5 With Division I Power (SL-73) Out-of-Service)

Problem Description

A Division I Power Outage is normally required for maintenance activities during each refueling outage. When Safety-Related bus SL-73 is taken out of service power is lost to the Reactor Water Cleanup System (RWCU) non-regenerative heat exchanger outlet temperature switch (RWCU-TIS-8). This, in turn, closes the outboard containment isolation valve (RWCU-V-4) which isolates the RWCU System. During refueling there is a need to keep RWCU operational to maintain water quality.

Discussion and Corrective Action

The Plant Procedure on Removing SL-73 from Service (PPM 2.7.14) was changed to allow the installation of a Lifted Lead and Jumper to deactivate the temperature switch (RWCU-TIS-8) if preferred during an outage.

A 50.59 evaluation was performed to support this change in plant procedures. The operation of the plant with this jumper in place would not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the Reactor Water Cleanup System can operate safely without the temperature switch during Modes 4 and 5, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PDF 289-0289 (Procedure Change to Allow Refuel Bridge Operation With Mode Switch In Shutdown)

Problem Description

The Reactor Manual Control System (RMCS) logic was not designed correctly and can send incorrect signals to the mode switch for the Refueling Bridge.

Discussion and Corrective Action

A procedure change (PPM 2.14.1) was processed to allow the installation of a jumper to defeat the bridge's logic input from RMCS on mode switch position. This will allow refuel bridge operation while the reactor mode switch is in shutdown. The jumper will simulate the reactor mode switch being in the refuel position thus allowing continued bridge operation over the core. Implementation of this change during the Spring 1989 refueling outage was not required. Implementation at any time in the future will require a Lifted Lead and Jumper.

A 50.59 evaluation was performed to support this change in plant procedures. The operation of the plant with this jumper in place would not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the refueling platform and its restrictions would not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PDF 289-0394 (Operation of the Residual Heat Removal "B" Pump Without the Suction Valve Full Open Interlock)

Problem Description

During the Spring 1989 refueling outage maintenance was being performed on the Limitorque Operator for the valve (RHR-V-9) that provides isolation to the suction of Residual Heat Removal Pump "B" (RHR-P-2B). RHR-V-9 has a full open interlock that prevents the pump from starting if the valve is not fully open and trips the pump when the valve starts to close. It was desired to have an additional shutdown cooling method available by using RHR loop "B" during this phase of the outage.

Discussion and Corrective Action

A 50.59 Safety Significance review was performed to install an electrical jumper and a temporary procedure deviation was approved to change the operating procedure (PPM 2.4.2) to allow operation of RHR-P-2B without the RHR-V-9 full open interlock during the short period of time while maintenance was being performed on the Limitorque operator on RHR-V-9.

This change did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the Residual Heat Removal System was administratively controlled and did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

ISCR-920 (Change to Reduce Temperature Range of Accident Monitoring Recorder)

Problem Description

The Residual Heat Removal (RHR) Heat Exchanger Outlet Temperature had a range of 0 to 600 degrees F. This recorder is part of the accident monitoring instrumentation and monitors the RHR and Fuel Pool Cooling temperatures. The readability of the chart recorder needed to be improved.

Discussion and Corrective Action

The recorder range was reduced to 0 to 500 degrees F. The reduced span provided better resolution in both the operating and accident temperature ranges.

This change did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the overall operation of the accident monitoring instrumentation did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PDF 289-0963 (Reactor Core Isolation Cooling Valves Inoperable/Bypass
PDF 289-0949 and Inoperable Status Indication Change)

Problem Description

Two Reactor Core Isolation Cooling Valves (RCIC-V-22 and RCIC-V-59) were taken out-of-service to effect repair of RCIC-V-22. These two valves are the test return to Condensate Storage Tank flow control and stop valves, respectively. Deenergizing and tagging out the valves caused the Bypass and Inoperable Status Indication (BISI) system to activate masking any signals that may be present on eleven other RCIC valves associated with annunciator 4.601.A4-6.8, "RCIC DIVISION I OUT-OF-SERVICE" and the "MOTOR OPERATED VALVE NETWORK POWER LOSS/OVERLOAD" BISI.

Discussion and Corrective Action

Relays RCIC-RLY-80/22 and RCIC-RLY-80/59 were pulled to allow the remaining motor operated valves to be monitored by the BISI system. Plant procedure PPM 4.601.A4-6.8 was deviated to show the change.

The operation of the plant with the BISI system modified as noted above did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question concluded: (1) the performance of the BISI System was improved, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

PDF 289-964 (Modification of the Plant Procedure on Wire Marker Installation)

Problem Description

The plant procedure (PPM 10.25.61) used to mark cables and wires in the plant required clarification on the practice of marking the wires which are broken out of a multi-conductor cable.

Discussion and Corrective Action

The plant procedure (PPM 10.25.61) was changed to allow the use of plastic sleeves or Brady markers which have black lettering on a white background as an approved installation method.

The operation of the plant with the wire marking procedure modified as noted above did not result in a change to the WNP-2 Technical Specifications and the unreviewed safety question concluded: (1) the wire marking criteria and performance was not changed, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.

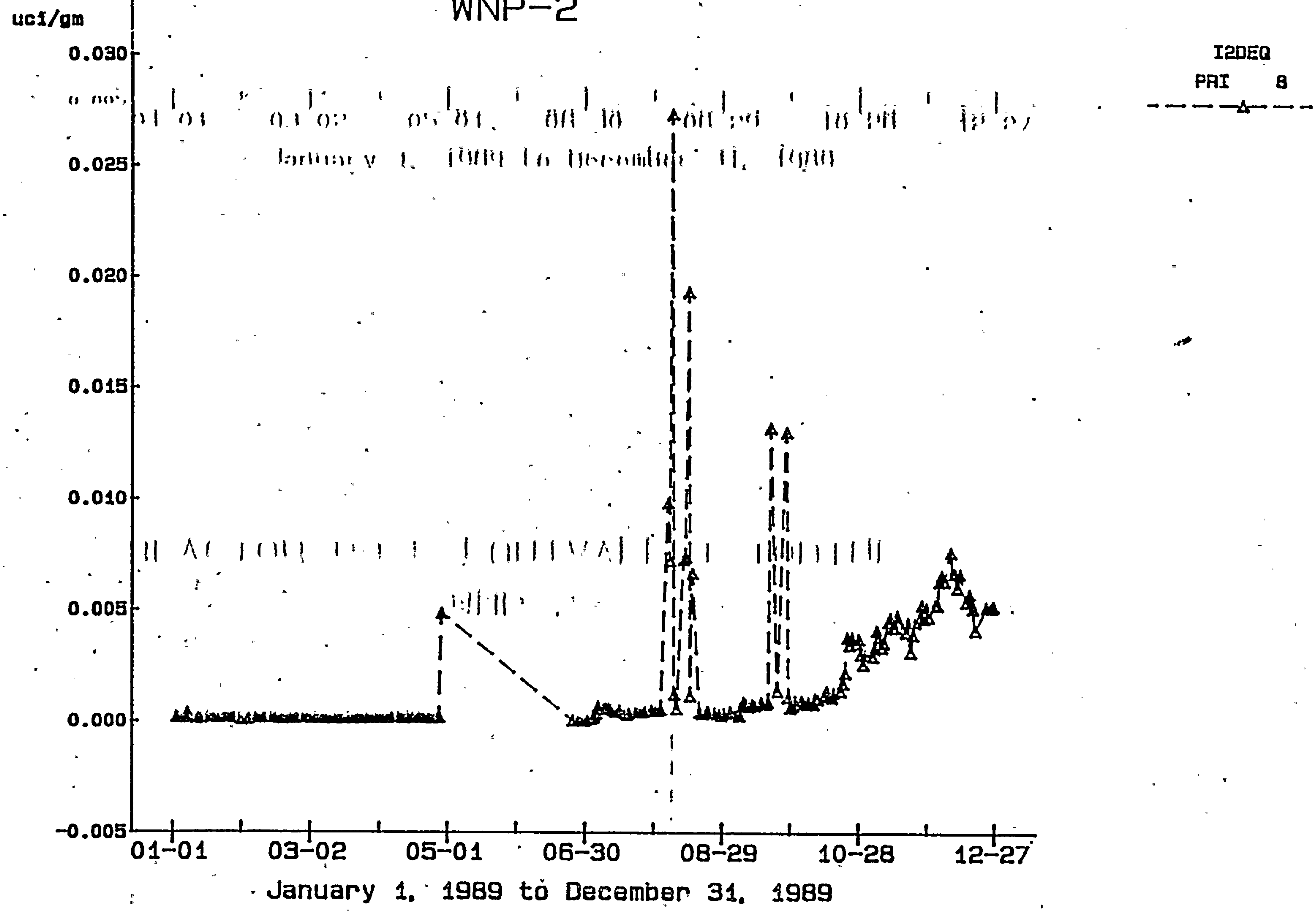
2.9 REACTOR COOLANT SPECIFIC ACTIVITY LEVELS

This section contains information relative to reactor coolant cumulative iodine levels, iodine spikes and specific activity of all isotopes other than iodine.

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 and paragraph 6.9.1.5.c., (see 1989 cumulative iodine graph, attached). The specific activity of the primary coolant was routinely sampled and analyzed as required by WNP-2 Technical Specifications, and was in all cases, less than or equal to 100/E microcuries per gram.

A graph showing cumulative iodine dose equivalent for the calendar year 1989 follows.

REACTOR DOSE EQUIVALENT IODINE
WNP-2



2.10 REPORT OF DIESEL GENERATOR FAILURES

This section contains information pertaining to the reporting of diesel generator failures, valid and nonvalid, in accordance with the requirements of WNP-2 Technical Specification 4.8.1.1.3. This report provides the information required by Regulatory Position C.3.b of Regulatory GUide 1.108, Revision 1, August 1977.

Diesel Generator Failure Number One

1. Identity of diesel generator unit and date of failure:

   Division Three Emergency Diesel Generator (DG-3)
   May 12, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure. The unit was inoperable for maintenance overhaul activity.

3. Cause of failure:

   The exact cause of the failure was not able to be determined. During a slow start test (nonvalid test) run of DG-3, the unit started at full speed (900 RPM) rather than slow speed (400 RPM) as designed. Subsequent tests were not able to duplicate the failure.

4. Corrective measures taken:

   None.

5. Length of time diesel generator unit was unavailable:

   Not applicable for this nonvalid failure.

6. Current surveillance test interval:

   Thirty-one days.

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Two

1. Identity of diesel generator unit and date of failure:

   Division Three Emergency Diesel Generator (DG-3)
   May 13, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure. The test was a nonvalid test because it was testing a feature which was not a part of the defined diesel generator unit design.

3. Cause of failure:

   The failure was the result of a nonvalid test performed to discover if DG-3 would start with one of two starting air headers isolated. The ability to start on one air header is not part of the DG-3 design. The cause of the failure to start was insufficient capacity of only one starting air header. The design of this system calls for two starting air headers.

4. Corrective measures taken:

   None.

5. Length of time diesel generator unit was unavailable:

   Not applicable. This was a nonvalid test.

6. Current surveillance test interval:

   Thirty-one days.

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Three

1. Identity of diesel generator unit and date of failure:

   Division One Emergency Diesel Generator (DG-1).
   May 18, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure. The unit was inoperable for maintenance overhaul activity.

3. Cause of failure:

   No definite cause of failure was able to be identified. The unit tripped during an end of maintenance warranty run prior to declaration of operability. A thorough investigation was unable to identify a definite cause. The fault trip was not able to be repeated during follow-up testing.

4. Corrective measures taken:

   The 18-month overhaul procedure was modified to include a specific check of the manual overspeed trip mechanism.

5. Length of time diesel generator unit was unavailable:

   Not applicable for this nonvalid failure.

6. Current surveillance test interval:

   Thirty-one days.

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Four

1. Identity of diesel generator unit and date of failure.

   Division Two Emergency Diesel Generator (DG-2)
   May 20, 1989

2. Number designation of failure in last 100 valid tests.

   Not applicable. This was a nonvalid test failure. Per WNP- 2 Technical Specification Table 4.8.1.1.2-1, with the exception of the semiannual fast start, no starting time requirements are required to meet the valid test requirements of NRC reg. Guide 1.108.

3. Cause of failure:

   During performance of the 18 Month Logic System Functional Test DG-2 Loss of Power Test, the diesel generator did not attain rated speed within ten seconds of receiving a start signal. The cause of the failure was originally isolated to a broken pneumatic boost line which supplies the Woodward speed governor unit. This prevented the start boost signal from being received by the actuator and would have resulted in a decrease in fuel supply to the diesel during fast start.

4. Corrective measures taken:

   The pneumatic line was repaired and the unit was retested. The retest did not demonstrate acceptable starting time. (See Diesel Generator Failure #5.)

5. Length of time diesel generator unit was unavailable:

   Not applicable for this nonvalid test.

6. Current surveillance test interval:

   Thirty-one days

7. Verification of test interval.

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Five

1. Identity of diesel generator unit and date of failure:

   Division Two Emergency Diesel Generator (DG-2)
   May 24, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid test failure. Per WNP- 2 Technical Specification Table 4.8.1.1.2-1, with the exception of the semiannual fast start, no starting time requirements are required to meet the valid test requirements of NRC reg. Guide 1.108.

3. Cause of failure:

   During performance of the 18 Month Logic System Functional Test DG-2 Loss of Power Test, the diesel generator did not attain rated speed within ten seconds of receiving a start signal. The cause of the failure was isolated to the voltage permissive relay DG-RLY-59/DG2 which provides a permissive signal to close the DG2 output breaker when generated voltage is high enough. The relay actuation setpoint calibration tolerance was found to allow sufficient variation to affect the 10 second start time under certain conditions.

4. Corrective measures taken:

   The relay was recalibrated to obtain sufficient setpoint performance to ensure obtaining a maximum 10 second start time. This mechanical relay was later replaced with a solid state relay which could be calibrated to perform consistently within the required tolerance.

5. Length of time diesel generator unit was unavailable:

   Not applicable. This was a nonvalid failure.

6. Current surveillance test interval:

   Thirty-one days.

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Six

1. Identity of diesel generator unit and date of failure:

   Division Three Emergency Diesel Generator (DG-3)
   June 2, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure as it was due to personnel error.

3. Cause of failure:

   During performance of the Logic System Functional Loss of Power Test, the diesel operator did not apply sufficient load soon enough after synchronization with the power grid to prevent a reverse power trip of the DG unit.

4. Corrective measures taken:

   The operating procedure was evaluated for correctness and found to be acceptable. The operator was counselled.

5. Length of time diesel generator unit was unavailable:

   DG3 was unavailable for approximately ten minutes while the protective relays were being reset.

6. Current surveillance test interval:

   Thirty-one days

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Seven

1. Identity of diesel generator unit and date of failure:

   Division One Emergency Diesel Generator (DG-1)
   June 9, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure. This was not a valid test failure. Per WNP-2 Technical Specification Table 4.8.1.1.2-1, with the exception of the semiannual fast start, no starting time requirements are required to meet the valid test requirements of NRC Reg. Guide 1.108.

3. Cause of failure:

   During performance of the 18 Month Logic System Functional Test DG-1 Loss of Power Test, the diesel generator did not attain rated speed within ten seconds of receiving a start signal. The cause of the failure was incorrect connection of the start pneumatic boost signal to the Woodward speed governor unit. This resulted in insufficient fuel supply to the diesel during fast start to ramp speed at the required rate.

4. Corrective measures taken:

   The pneumatic line was connected to the correct port on the governor actuator. The unit was then retested successfully. The other diesel units were inspected for similar fault.

5. Length of time diesel generator unit was unavailable:

   Not applicable to this nonvalid start.

6. Current surveillance test interval:

   Thirty-one days

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

Diesel Generator Failure Number Eight

1. Identity of diesel generator unit and date of failure:

   Division One Emergency Diesel Generator (DG-1)
   June 10, 1989

2. Number designation of failure in last 100 valid tests:

   Not applicable. This was a nonvalid failure. The failed valve is not a part of the defined diesel generator unit design.

3. Cause of failure:

   The DG-1 output circuit breaker was tripped by protective relay actuated by high engine cooling water temperature. This, in turn, shutdown the diesel generator. The high engine cooling water temperature was caused by failure of the Standby Service Water System cooling water inlet valve which supplies cooling water to the engine cooling water heat exchanger. The valve disk separated from the operating stem and remained in the closed position blocking cooling water flow. The pneumatic valve operator, however, stroked fully open showing a full open valve position indication.

4. Corrective measures taken:

   The faulty valve was removed. The possible generic implications of this failure were investigated. These valves on the remaining Diesel Generator units were removed. (See PER 289-0487.)

5. Length of time diesel generator unit was unavailable:

   Not applicable to this nonvalid failure.

6. Current surveillance test interval:

   Thirty-one days

7. Verification of test interval:

   The surveillance test interval of thirty-one days is in conformance with NRC Reg. Guide 1.108 position C.2.d.

2.11 FIRE PROTECTION PROGRAM CHANGES

The following changes were made to the fire protection program during the calendar year. These revisions were all made to plant procedure 1.3.10, Fire Protection Program, in which the procedure is included as part of the FSAR by reference.

1. The procedure was revised to require all detectors in vital areas to be operational at all times. If they are not operational compensatory measures must be taken. The revision is more restrictive than the previous requirements.

2. Detector maintenance activities were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

3. The minimum requirements for fire protection system pump and water supply operability were clarified. The requirement is: two 2000 gpm pumps and the 2500 gpm pump must be operable at all times.

4. Fire Suppression Water System Maintenance activity descriptions were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Fire Suppression Water System Maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

5. Operability requirements for hydrants within the protected area were changed to require operability of only those hydrants that provide protection for equipment that is required to be operable. This is more restrictive than the previous requirement.

6. The compensatory actions associated with an impaired hydrant or hose house were changed in that a hose is now required to be placed at an adjacent hydrant/hose house in 24 hours instead of one hour. The basis for the change is that a van is used by the fire brigade when responding to fires in the protected area. The van has hose on board and can be used to rapidly lay the hose for fire fighting.

7. Maintenance activities associated with hydrants were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

8. Operability requirements for hose stations were changed to require all hose stations in the Corridors, Turbine, Reactor, Radwaste and Diesel Buildings be operable anytime the equipment which the hydrant provides protection for is required to be operable. This is more restrictive than the previous requirement.

9. Hose station maintenance activity descriptions were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Hose station maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

10. The requirements for a fire watch to be stationed for inoperable control room halon systems was removed since the control room is manned 24 hours per day and there are halon extinguishers available. A fire protection impairment must be issued.

11. Halon fire protection maintenance activity descriptions were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Halon fire protection maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

12. The requirements for compensatory measures associated with suppression systems were changed to require a fire impairment permit. The requirement of a fire watch is needed only if a system is inoperable and the associated detection system is inoperable. If the detection system is operable an hourly fire tour must be established.

13. Various valve and fire protection equipment maintenance activity descriptions were removed from plant procedure 1.3.10 and moved to volume 15 procedures. Valve and fire protection maintenance activities will be performed in accordance with the applicable NFPA standards, as well as insurance company and manufacturer recommendations. Maintenance activities will be scheduled via the Scheduled Maintenance System (SMS).

14. The requirements for various inspection activities were changed to be in compliance with NFPA standards and insurance company and manufacturer recommendations. It additionally states that the maintenance will be scheduled via the Scheduled Maintenance System (SMS).

The modifications to the Fire Protection Program as described above did not result in a change to the WNP-2 Technical Specifications since the Fire Protection Technical Specifications had been removed per Amendment 67. The unreviewed safety question evaluation concluded: (1) the function and performance of the Fire Protection Program did not change, (2) the margin of safety provided in the technical specifications was not changed, and (3) the boundary conditions for the FSAR evaluations were not changed.