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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

Docket No. 50-397

March 7, 1991

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Martin:

Subject: **NUCLEAR PLANT NO. 2 ANNUAL OPERATING REPORT 1990**

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 Operation Information Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Operating Report for calendar year 1990. Should you have any questions or comments, please contact G. L. Gelhaus, WNP-2 Assistant Plant Technical Manager.

Very truly yours,

JWBaker

J. W. Baker
WNP-2 Plant Manager

JWB:CLF:dm
Attachments

cc: Mr. John B. Martin, NRC Region V
Mr. C. Sorenson, NRC Resident Inspector (M/D 901A)
Mr. D. L. Williams, BPA (M/D 399)
Mr. R. F. Mazurkiewicz, BPA (M/D 399)

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Forwards Addl Info re Station Blackout, Per 910425 Telcon w/ NRC & Science Applications Intl Corp.. Dated May 7, 1991.

431-97-0141 box #8

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ANNUAL OPERATING REPORT

OF

WNP-2

FOR 1990

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

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1.0 INTRODUCTION

The 1990 Annual Operating Report of Washington Public Power Supply System Plant Number 2 (WNP-2) is submitted in accordance with the requirements of Federal Regulations and Facility Operating License NPF-21. Plant WNP-2 is a 3323 MWt, BWR-5, which began commercial operation on December 13, 1984.

Following a record 203 consecutive days of operation, the Plant was shutdown on April 21, 1990 for the annual maintenance and refueling outage. The outage had originally been scheduled for 45 days; however, on May 27, 1990 the Division 1 Emergency Diesel Generator failed approximately six hours into a 24-hour, full-load run due to the failure of the diesel generator slip ring end bearing. During an investigation into this event it was also discovered that both the Division 1 and Division 2 Emergency Diesel Generators had shorted field pole windings which were evaluated as being the result of manufacturing defects. The Division 1 Diesel Generator had to be completely rewound, resulting in a seven-week extension of the outage. Following successful Diesel Generator repair and testing efforts, the Plant was restarted on August 4, 1990 and operated until September 25, 1990 when the reactor was manually scrammed after experiencing main turbine hydraulic control oil pressure problems in the Digital Electro-Hydraulic (DEH) System. The oil pressure problems were caused by a broken pipe nipple in the auto-stop oil header portion of the Turbine Lube Oil System. Repairs were made and the Plant was restarted on September 30, 1990 following a five-day outage.

On November 2, 1990 the Plant was manually shutdown after confirmation by Nondestructive Examination (NDE) testing of a small crack in a 3/4-inch drain line off of the High Pressure Core Spray (HPCS) System injection header. The crack was repaired and NDE testing was performed on 104 welds on similar drains in the Emergency Core Cooling System (ECCS) prior to returning the Plant to service on November 11, 1990. On December 7, 1990 the Main Generator tripped at 100 percent power following flashover-to-ground on a "B" phase high voltage insulator between the main step-up transformers and generator disconnects. The electrical fault (flashover) was due to Circulating Water (CW) System cooling tower chemical deposits having built up on the insulator, with wet and icing conditions contributing to provide a conductive path over the surface of the insulator. The damaged insulator stack was replaced and all other 500 KV, 230 KV and 115 KV insulators in the main transformer yard were inspected and cleaned. The Plant was restarted on December 9, 1990 and ran at or near 100 percent capacity for the remainder of the year.

During 1990, 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 fifth refueling outage was successfully completed. Significant activities included:

- o Preventive maintenance on the remaining four Main Steam Isolation Valves (MSIVs).
- o Bearing repair and complete rewind of the Division I Emergency Diesel Generator.
- o Inspection of two of the three Emergency Diesel Generators and overhauling the High Pressure Core Spray (HPCS) diesel engine.
- o Inspection of one of three Low-Pressure Turbine Rotors. Non-destructive examination of the rotor confirmed crack indications and four blades were replaced.
- o Preventative Maintenance on 30 Control Rod Drive Mechanisms (CRDMs). This activity included removing, replacing and rebuilding the CRDMs.
- o Replacement of the rubber seals between the low-pressure turbines and the condenser boxes to minimize air leakage into the condenser.

- o Removal of spent fuel assemblies and refueling the reactor. The refueling activity included replacing 152 fuel assemblies, using a fuel shuffle scheme.

(b) The best operating cycle of 203 days of continuous operation ended when the Plant was shutdown for the annual maintenance and refueling outage. During this cycle 6.5 billion net kilowatt-hours were generated, enough electricity to supply the annual needs of more than 400,000 all-electric homes. Furthermore, the capacity factor for the operating cycle was more than 84 percent.

(c) During October, 1990 a new monthly generation record was set when the Plant provided 787,321 megawatt-hours of electricity to the Bonneville Power Administration's regional transmission system. This compares to the old plant record of 780,000 megawatt-hours generated during December, 1989.

In 1990 total radiation exposure at the Plant was 535 man-rem, as compared to the 1989 level of 492 man-rem. (The Institute for Nuclear Power Operation (INPO) had set 460 man-rem as the 1990 industry goal for BWRs.)

During the year WNP-2 received 15 NRC Notices of Violation (NOVs): One (1) Level II, one (1) Level III, twelve (12) Level IV and one (1) Level V. The Level II violation was associated with previous (1986) fire protection issues for which no response was required, nor was a civil penalty proposed. The Level III violation was associated with previous (1986) equipment qualification issues pertaining to splices in containment and included a proposed $50,000 civil penalty.

Also during 1990, a total of 32 Licensee Event Reports (LERs) were written and submitted pursuant to the requirements of 10CFR50.73, as compared to a total of 45 LERs submitted during 1989.

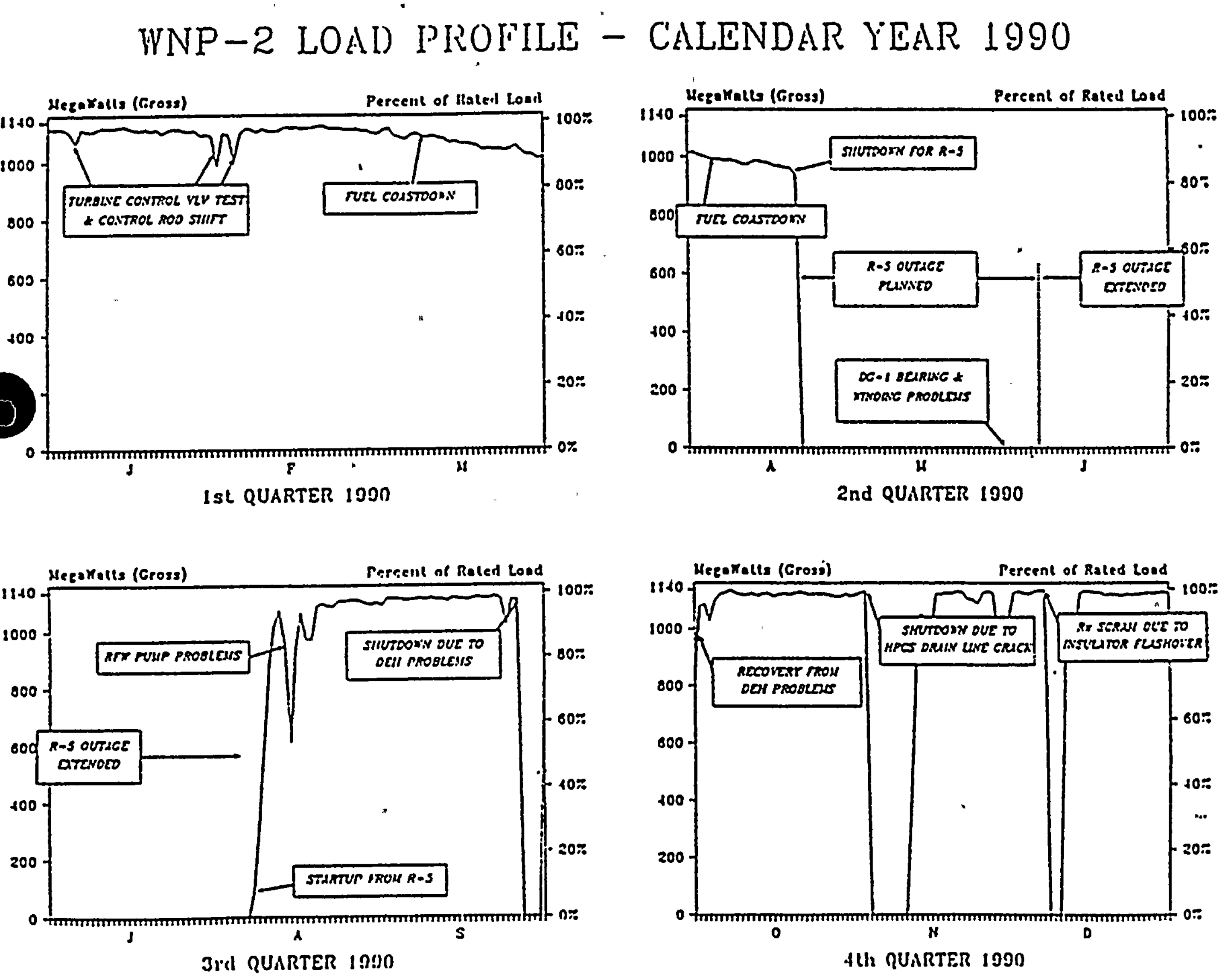
The 1990 capacity factors, based upon net electrical energy output, are listed in the following table.

Month	Capacity Factor
January	93.59
February	93.77
March	91.87
April *	55.80
May	0
June	0
July **	0
August	67.12
September	78.24
October	96.51
November	62.15
December	84.14
Overall	59.86

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

1.1 1990 WNP-2 LOAD PROFILE

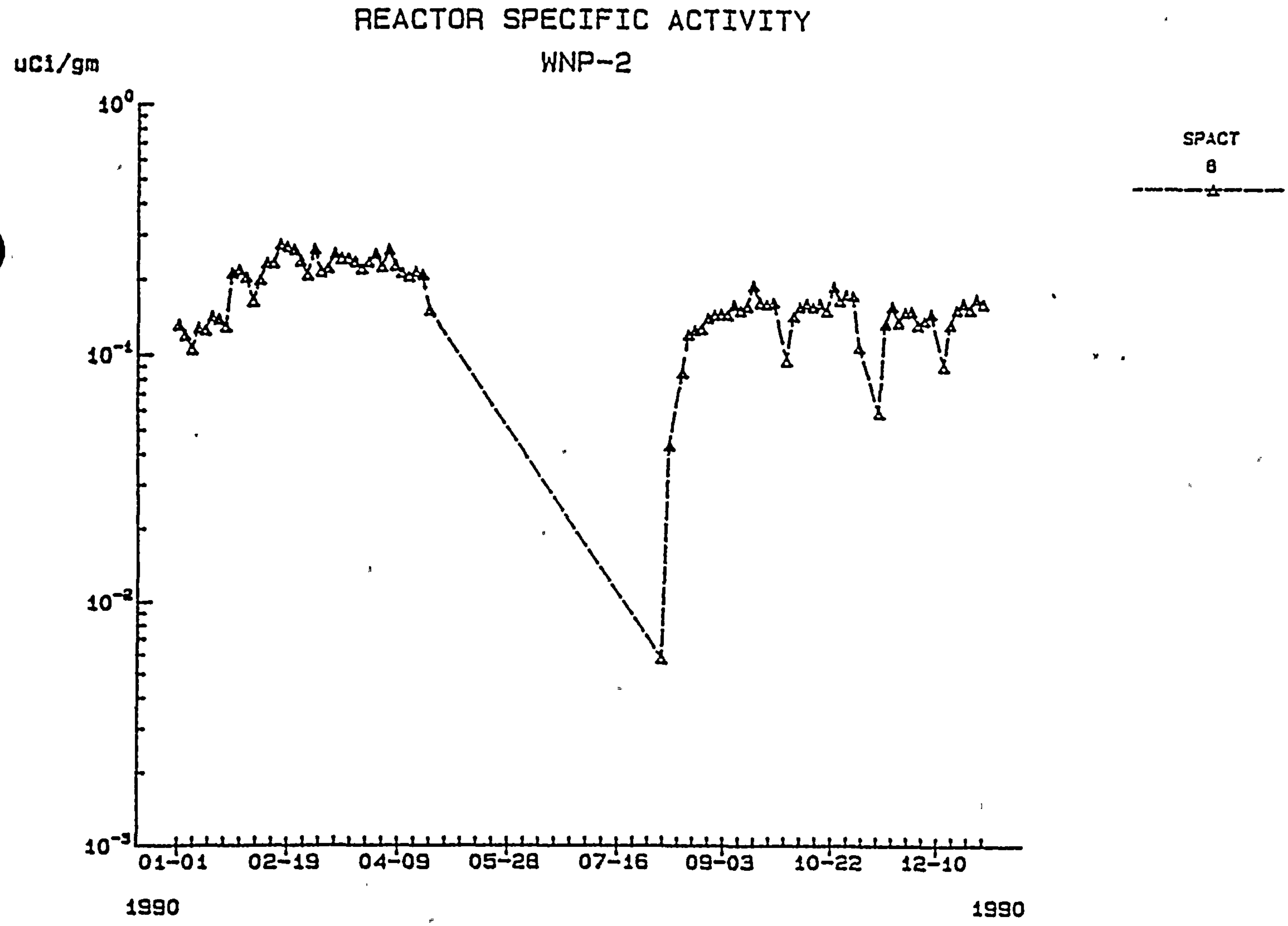
The 1990 Power History graph for WNP-2 is shown below.



1.2 REACTOR COOLANT SPECIFIC ACTIVITY LEVELS

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

The specific activity of the primary coolant was significantly less than 0.2 microcuries per gram dose equivalent I-131 as set forth in WNP-2 Technical Specification LCO 3/4.4.5. 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.



2.0 REPORTS

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

2.1 ANNUAL PERSONNEL EXPOSURE AND MONITORING REPORT

The information provided in this section of the report is required by the WNP-2 Technical Specifications, Section 6.9.1.5.a, and Regulatory Guide 1.16, Revision 4.

RER-020 RADIATION EXPOSURE RECORDS 02/08/91 10:47
WORK AND JOB FUNCTION REPORT / 1.16 APPENDIX A

NUCLEAR PLANT NO. 2 REPORT FOR CALENDAR YEAR 1990

		NUMBER OF PERSONS RECEIVING OVER 100 MREM			TOTAL MAN-REM		
		STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTOR AND OTHERS
OPERATIONS & SURVEILLANCE	MAINTENANCE PERSONNEL	42.685	0.808	12.135	28.336	0.274	9.008
	OPERATING PERSONNEL	47.684	1.086	0.000	42.304	0.163	0.000
	HEALTH PHYSICS PERSONNEL	29.777	0.000	15.458	20.280	0.000	9.798
	SUPERVISORY PERSONNEL	12.069	0.964	0.000	4.570	0.231	0.000
	ENGINEERING PERSONNEL	4.839	9.411	2.053	1.104	3.142	0.525
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	174.403	0.178	186.685	141.120	0.060	106.110
	OPERATING PERSONNEL	2.713	0.086	0.000	2.795	0.014	0.000
	HEALTH PHYSICS PERSONNEL	8.765	0.000	41.595	10.130	0.000	30.589
	SUPERVISORY PERSONNEL	13.174	1.594	2.297	6.147	0.586	0.498
	ENGINEERING PERSONNEL	20.576	17.948	26.804	7.671	6.607	8.894
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	0.573	0.000	2.921	0.638	0.000	2.164
	OPERATING PERSONNEL	0.008	0.000	0.000	0.009	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.253	0.000	0.419	0.409	0.000	0.363
	SUPERVISORY PERSONNEL	0.077	0.511	0.180	0.023	0.163	0.051
	ENGINEERING PERSONNEL	1.623	1.053	9.100	0.479	0.553	2.133
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	15.158	0.014	65.247	16.121	0.005	24.446
	OPERATING PERSONNEL	0.068	0.000	0.000	0.058	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.977	0.000	3.059	1.106	0.000	2.252
	SUPERVISORY PERSONNEL	0.949	0.931	0.523	0.814	0.191	0.149
	ENGINEERING PERSONNEL	1.031	3.666	5.688	0.367	1.147	1.322
WASTE PROCESSING	MAINTENANCE PERSONNEL	8.580	0.000	3.676	5.905	0.000	1.562
	OPERATING PERSONNEL	0.112	0.000	0.000	0.164	0.000	0.000
	HEALTH PHYSICS PERSONNEL	3.096	0.000	1.902	2.799	0.000	3.181
	SUPERVISORY PERSONNEL	0.041	0.000	0.000	0.037	0.000	0.000
	ENGINEERING PERSONNEL	0.092	0.316	0.107	0.037	0.046	0.033
REFUELING	MAINTENANCE PERSONNEL	2.424	0.000	0.018	1.654	0.000	0.009
	OPERATING PERSONNEL	2.516	0.000	0.000	3.817	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.678	0.000	0.016	1.431	0.000	0.019
	SUPERVISORY PERSONNEL	1.031	0.000	0.000	0.270	0.000	0.000
	ENGINEERING PERSONNEL	0.000	0.076	0.496	0.000	0.019	0.074
TOTAL	MAINTENANCE PERSONNEL	243.823	1.000	270.682	193.774	0.339	143.299
	OPERATING PERSONNEL	53.101	1.172	0.000	49.147	0.177	0.000
	HEALTH PHYSICS PERSONNEL	43.546	0.000	62.449	36.155	0.000	46.202
	SUPERVISORY PERSONNEL	27.341	4.000	3.000	11.861	1.171	0.698
	ENGINEERING PERSONNEL	28.161	32.470	44.248	9.653	11.514	12.981
GRAND TOTAL		395.972	38.642	380.379	300.595	13.201	203.180

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information concerning main steam line safety/relief valve (SRV) challenges for calendar year 1990 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

There were no SRV actuations in the first quarter of 1990.

Second Quarter

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)*	REASON FOR ACTUATION (CODE)*	PRIOR PLANT CONDITIONS (CODE)*	POWER LEVEL %	ASSOCIATED LER
04/21/90	MS-RV-1A	B	C	D	15	--
04/21/90	MS-RV-2A	B	C	D	14	--
04/21/90	MS-RV-3A	B	C	D	15	--
04/21/90	MS-RV-4A	B	C	D	15	--
04/21/90	MS-RV-1B	B	C	D	14	--
04/21/90	MS-RV-2B	B	C	D	15	--
04/21/90	MS-RV-3B	B	C	D	15	--
04/21/90	MS-RV-4B	B	C	D	15	--
04/21/90	MS-RV-5B	B	C	D	14	--
04/21/90	MS-RV-1C	B	C	D	15	--
04/21/90	MS-RV-2C	B	C	D	14	--
04/21/90	MS-RV-3C	B	C	D	15	--
04/21/90	MS-RV-4C	B	C	D	14	--
04/21/90	MS-RV-5C	B	C	D	15	--
04/21/90	MS-RV-1D	B	C	D	14	--
04/21/90	MS-RV-2D	**				
04/21/90	MS-RV-3D	B	C	D	15	--
04/21/90	MS-RV-4D	B	C	D	15	--

These actuations were performed to test acoustic monitors.

04/21/90	MS-RV-1A	C	C	D	0	--
04/21/90	MS-RV-2A	C	C	D	0	--
04/21/90	MS-RV-4A	C	C	D	0	--
04/21/90	MS-RV-1C	C	C	D	0	--
04/21/90	MS-RV-3C	C	C	D	0	--
04/21/90	MS-RV-4C	C	C	D	0	--
04/21/90	MS-RV-5C	C	C	D	0	--
04/21/90	MS-RV-1D	C	C	D	0	--
04/21/90	MS-RV-3D	C	C	D	0	--
04/21/90	MS-RV-4D	C	C	D	0	--

These actuations involved "simmering" the valves for in-situ setpoint verification testing. SRV-2A was "simmered" six times; SRV-1D, four times; all other SRVs, two times.

* Codes are explained on page 9.
** 2D manual operator not functional. Acoustic monitor verified on 8/6/90 prior to resuming operations.

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

Third Quarter

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)*	REASON FOR ACTUATION (CODE)*	PRIOR PLANT CONDITIONS (CODE)*	POWER LEVEL %	ASSOCIATED LER
08/05/90	MS-RV-1A	C	C	C	1.5	--
08/05/90	MS-RV-2A	C	C	C	1.5	--
08/05/90	MS-RV-3A	C	C	C	1.5	--
08/05/90	MS-RV-3B	C	C	C	1.5	--
08/05/90	MS-RV-4B	C	C	C	1.5	--
08/05/90	MS-RV-5B	C	C	C	1.5	--

These actuations involved "simmering" the valves for in-situ setpoint verification testing. Each valve "simmered" two times.

08/06/90	MS-RV-1A	B	C	C	15	--
08/06/90	MS-RV-2A	B	C	C	15	--
08/06/90	MS-RV-3A	B	C	C	15	--
08/06/90	MS-RV-4A	B	C	C	15	--
08/06/90	MS-RV-1B	B	C	C	15	--
08/06/90	MS-RV-2B	B	C	C	15	--
08/06/90	MS-RV-3B	B	C	C	15	--
08/06/90	MS-RV-4B	B	C	C	15	--
08/06/90	MS-RV-5B	B	C	C	15	--
08/06/90	MS-RV-1C	B	C	C	15	--
08/06/90	MS-RV-2C	B	C	C	15	--
08/06/90	MS-RV-3C	B	C	C	15	--
08/06/90	MS-RV-4C	B	C	C	15	--
08/06/90	MS-RV-5C	B	C	C	15	--
08/06/90	MS-RV-1D	B	C	C	15	--
08/06/90	MS-RV-2D	B	C	C	15	--
08/06/90	MS-RV-3D	B	C	C	15	--
08/06/90	MS-RV-4D	B	C	C	15	--
08/07/90	MS-RV-5B	B	C	C	15	--
09/29/90	MS-RV-1B	B	C	C	15	--

These actuations were performed to test acoustic monitors.

Fourth Quarter

12/07/90	MS-RV-1B	A	A	E	98	90-031

This actuation was in response to a unit trip.

* Codes are explained on page 9.

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

CODES:

Type of Actuation

A. Automatic
B. Remote Manual
C. Spring

Plant Condition

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

Reason for Actuation

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

NOTES:

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

2) Spring set testing was performed in accordance with ASME Section XI and Technical Specification 4.0.5 requirements.

2.3 SUMMARY OF PLANT OPERATION

This section of the report responds to the requirements of Regulatory Guide 1.16, Revision 4, Section C.1.b. Major safety-related corrective maintenance which is covered in Section 2.4.

2.3 SUMMARY OF PLANT OPERATION

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE (CODE)	SHUT DOWN (CODE)	LER NUMBER	SYSTEM	COMPONENT	CAUSE & ACTION TO PREVENT RECURRENCE
1/30/90	S	0	H	5	--	RB	CONROD	Reduced power to perform a control rod sequence exchange.
4/21/90	S	2591.1	C	1	--	RC	FUELXX	Plant shutdown as scheduled for refueling outage R-5. Outage extended due to the failure of a Diesel Generator.
8/7/90	S	4.55	B	1	--	HA	MECFUN	Generator was removed from grid to perform overspeed testing of main turbine. It was then returned to service after successful completion of overspeed tests.
9/22/90	S	0	H	5	--	RB	CONROD	Reduced power to perform a control rod sequence exchange.
9/25/90	F	117.3	A	2	90-021	HA	TURBIN	Plant was forced down by low DEH control system pressure. A broken nipple on autostop hydraulic oil system resulted in decreasing DEH pressure which forced a rapid downpower with a manual scram being initiated at 40% power. Repairs were made and the hydraulic system tested prior to returning unit to service.

2.3 SUMMARY OF PLANT OPERATION (Continued)

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE (CODE)	SHUT DOWN (CODE)	LER NUMBER	SYSTEM	COMPONENT	CAUSE & ACTION TO PREVENT RECURRENCE
11/2/90	F	208.1	A	1	90-028	SF	PIPEXX	The plant was shut down after confirmation by NDE testing of a crack in a 3/4" drain line off HPCS injection header. The crack was repaired and NDE testing was performed on 104 welds on similar drains in the ECCS system prior to returning plant to service.
11/28/90	F	0	A	5	--	CH	INSTRU	Power was reduced due to feedwater level control difficulties caused by valve linkage problems.
12/7/90	F	79.7	A	3	90-031	EB	ELECON	Generator tripped at 100% power following flashover to ground on a "B" phase high voltage insulator between the main stepup transformers and generator disconnects. The damaged insulator was replaced and the remaining insulators were cleaned and inspected prior to restart.

2.3 SUMMARY OF PLANT OPERATION (Continued)

CAUSE CODE	TOTAL FOR 1989	TOTAL GENERATOR OFF-LINE HOURS
A	4	405.1
B	1	4.55
C	1	2591.1
D	0	0
F	0	0
G	0	0
H	2	0
	TOTAL	3000.75

2.3 SUMMARY OF PLANT OPERATION (Continued)

SUMMARY OF CODES

OUTAGE TYPE	CAUSE CODE	SYSTEM SHUTDOWN METHOD	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
			SF	Emergency Core Cooling System & Controls

2.3 SUMMARY OF PLANT OPERATION (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

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Mechanical Function Units (MECFUN)	Mechanical Controllers Governors Gear Boxes Varidrives Couplings
Pipes, Fittings (PIPEXX)	Pipes Fittings
Turbines (TURBIN)	Steam Turbines Gas Turbines Hydro Turbines
Codes Not Applicable (ZZZZZ)	

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

This section of the report is provided in accordance with the requirements of Regulatory Guide 1.16, Revision 4, Section C.1.b(2)(e).

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DSA-C-2C	HPCS Power - Diesel Starting Air	With the plant in a scheduled refueling outage, operators performing the monthly operability surveillance on the high pressure core spray diesel noted the air compressor for the diesel starting air would not load to sufficient pressure.	The hydraulic unloader assembly was found installed with the ports 180 degrees out of alignment.	Hydraulic unloader assembly was replaced with new same type assembly and operability was functionally verified.
MS-42-8BA6D	Main Steam	With the plant in its annual refueling outage, main control room operators attempting to open the main steam line inboard drain valve to main condenser using the handswitch observed the valve would not open. Valve would not have opened remotely, but manual operation was still available. This caused the main steam line to be degraded. The plant was not affected.	Fuses in the motor operator starter cubicle had blown. Electricians meggered, took stroke time and amperage readings after installing new fuses, but were unable to determine the cause of the blown fuses.	Replaced the blown fuses and took data with inconclusive results. No further work performed at this time.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RCIC-42-S11D3C .	Reactor Core Isolation Cooling	With the plant in a scheduled refueling outage, operators were performing valve lineups as required to perform monthly surveillance on the reactor core isolation cooling system. While attempting to stroke the valve for alternative pool supply from open to close position, the control fuses blew. The fuses were replaced once, but again blew while trying to close the valve from the control room.	Coil on the close contactor did not pass resistance and continuity checks. Cause of bad coil unknown. This resulted in loss of the train supplying the alternate water source for the suppression pool, but there was no effect on the plant.	Replaced bad coil with same type new coil. Verified proper valve operation.
RCIC-42-S11D3C	Reactor Core Isolation Cooling	With the plant in a scheduled refueling outage, the main control room received a call reporting smoke coming from a motor control center. Operators responded and noted the cubicle was for reactor core isolation cooling valve 10, which had just been stroked under the 28-day operability surveillance.	Plant electricians found the 125V DC coil on the right hand contactor had burned. Cause of burned coil was unknown.	Coil was replaced with same type new coil and valve was stroked to verify operability.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RPS-42-MG2/MS1	Reactor Protection	Plant was in Mode 5 (Refueling) during scheduled outage, when the reactor protection system motor generator set "B" tripped on starting as indicated by a main control room alarm. Plant was in a Division II scheduled outage, and when returning Division II to service the reactor protection system "B" served by Division II tripped causing a loss of Train B.	The overload heater block was found burned. The cause of the burned overload heater block was unknown.	Replaced the overload heater block and the overload relay. Placed the assembly into service and verified proper operability.
RPS-EPA-3F	Reactor Protection	Plant in Mode 5 for refueling outage, control room received an alarm on shutdown cooling isolation due to an overvoltage trip on the reactor protective system electrical protective assembly channel "F". Operators reestablished shutdown cooling, but received two subsequent isolations after which the breaker would not reclose.	Failure was traced to the circuit board (piece part of the circuit breaker) which was removed and returned to General Electric (GE) Nuclear Energy Division for failure analysis. GE subsequently reported the cause of failure was an engineering defect in the circuit board low frequency time delay function.	The circuit board and circuit breaker were replaced with new, like components. The assembly was calibrated and declared operable. Circuit boards will be modified to the revised GE design.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RWCU-42-S21A4B	Nuclear Steam Supply Shutoff (NSSS)	During valve lineups prior to functional testing of the reactor water cleanup system performed during the annual refueling outage, operators were attempting to open the outboard containment isolation valve using its handswitch in the main control room. The valve would not open using the handswitch.	Previous repair of thermal blocks in the motor control center done earlier in the outage had replaced the wiring for the control switch operation incorrectly.	Relanded wires correctly at the motor control center thermal block. Verified valve operation from the control room handswitch.
CRD-HCU-1043	Control Rod Drive	During scheduled refueling outage, the main control room received "Accumulator Trouble" alarm on Accumulator 10-43 which would not reset. Operators investigated and found nitrogen leakage at the accumulator cap connector. Attempts to tighten the connector were unsuccessful in stopping leakage.	Connector had worn out due to overuse/overtightening.	Replaced worn out connector with same type new connector. Pressurized the instrument block and "Snoop" tested for leakage.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
CRD-HCU-1839	Control Rod Drive	Plant was in scheduled refueling outage when the main control room received an "Accumulator Trouble" alarm on Accumulator 18-39. Operators investigated and found leakage on the accumulator connector. Attempts to tighten the connector were unsuccessful in stopping the leak.	Connector had worn out due to overuse and/or over-tightening.	Replaced the worn out connector with same type new connector. Assembly was then pressurized and "Snoop" tested for leakage with no leakage evident.
DG-ENG-1B1	Emergency Power	Plant in full power operation. During performance of monthly operability surveillance of the "B" emergency power diesel engines, the engines tripped on apparent overspeed signal with no indication of actual overspeed condition.	The setscrew which holds the trip lever arm in proper position had not been tightened sufficiently causing the arm to slip on the shaft to the point where it initiated the trip signal. Cause of loose setscrew thought to be due to initial installation done prior to plant startup.	Readjusted the trip lever arm and tightened the setscrew properly. Checked all other engines for proper setscrew installation.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DG-ENG-DG1	Emergency Power	During the 24 hour, full-load run on the Division I emergency diesel generator (performed during the outage to fulfill Technical Specification requirements), the generator was tripped by control room operators due to excessive bearing temperature and subsequent fire. The fire was extinguished and an unusual event was declared at 1810 hours. System redundancy was lost.	Failure analysis results concluded that the thrust bearing failed due to loss of lubrication caused by leakage of oil from the bearing oil reservoir. Leakage was caused by an inadequate o-ring seal as a result of an extra o-ring groove machined into the bearing which prevented a tight seal. This extra groove did not appear on any design drawings and is considered a manufacturing defect.	The generator was repaired at an offsite repair shop. Repair work consisted of bearing replacement and rewinding. The generator was reinstalled and operationally tested. Since the plant was already in an outage, the plant effect was that the outage had to be extended due to Technical Specification requirements for two diesel generators operable during power operations.
DLO-HX-2B2	Diesel Lube Oil	During scheduled maintenance performed on the Division I emergency diesel during scheduled refueling outage, a diesel lube oil leak at the piping exiting the diesel lube oil heat exchanger was observed.	O-ring seal had failed, most likely due to wear.	Disassembled the flexible connection and removed the o-ring. Replaced the o-ring with same type new o-ring. Reassembled connection and verified no visible leakage.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DCW-HX-1C	HPCS Power - Diesel Cooling Water	With the plant in a scheduled refueling outage and the high pressure core spray diesel in maintenance mode, operators noted that the diesel cooling water expansion tank required filling at least twice for the previous three days. They also noted the oil level in the engine sump had increased in the same time frame. A leak in the diesel cooling water to oil system was suspected.	Inspection of the tube sheet showed that the joint apparently had been assembled without packing between the tube sheet and shell joint. No visible damage of the tube sheet was evident.	Tube sheet was reassembled with new flange gasket and packing. A satisfactory system leakage test was then performed.
DSA-PS-3B	Diesel Starting Air	Plant was in a scheduled refueling outage. During the performance of biannual calibration, the diesel starting air pressure switch for the #2 compressor failed calibration. The switch would not close on decreasing pressure.	Contacts for decreasing pressure signal were degraded and unable to allow circuit completion. Cause of bad contacts was attributed to wear-out.	A spare set of contacts on the same switch were utilized to restore the pressure switch to the original configuration. Calibration was then performed successfully.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DO-LS-21	HPCS Power - Diesel Fuel Oil	Plant was in Mode 5, with the mode switch locked in refuel. During performance of biweekly diesel day tank operability surveillance, operators noted that the level switch (to start the transfer pump on low oil level) was not turning on the pump with low level condition in the tank.	Instrument and control technicians were unable to repeat the failure; however, they did find and remove two pieces of broken terminal block within the instrument housing. No other evidence of failure or damage was found.	Technicians removed the broken pieces of terminal block and verified the switch cycled properly through three transfer functions.
RFW-E/P-15	Feedwater	With the plant in a scheduled refueling outage, technicians were performing the 24 month surveillance on the current to pneumatic converter to feedwater flow control valve "15". The converter would not operate repeatedly.	The root cause was unknown.	The instrument was replaced with a same type new instrument. The calibration was then performed successfully.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RFW-E/P-2A	Feedwater	With the plant in its annual refueling outage, Instrument and Control (I&C) technicians were performing the 24 month calibration on the reactor feed pump 1A discharge electro/pneumatic converter flow signal to the temperature control valve. The technicians were unable to calibrate the unit due to continued drifting.	Cause of drifting problem was unknown.	Removed defective unit from service and replaced with a new same type unit. The calibration was then successfully completed.
MS-RIS-601B	Reactor Protection	Plant operating at full power. During performance of monthly surveillance test, main steam line radiation indicating switch "B" channel was found with the setpoint out of acceptance range and, during attempts to recalibrate, it was drifting excessively.	Exact cause unknown. These units have a history of problems, suspect end-of-life aging or wear-out.	Replaced the drawer unit with a spare same type drawer and finished performance of surveillance test satisfactorily.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-RIS-610D	Reactor Protection	Plant Operating at 100% power. Main control room operators noted spurious downscale trips on the "D" channel main steam line radiation monitor.	Voltage readings indicated signal input cable had loose shield connection at "J1" plug connector. Cause of loose connection was unknown.	Replaced "J1" connector with same type new connector. Placed unit into service and observed normal operation
RHR-PT-15B	Residual Heat Removal/Low Press Injection	During normal operation, an annual surveillance procedure was performed on the loop "B" residual heat removal flow transmitter. It showed a very sluggish response from approximately 15 mA/DC to 20 mA/DC. It took the unit approximately 16 minutes to change by 5 mA/DC. System function was not affected because the transmitter only provides indication for accident monitoring response.	The transmitter exhibited the same symptoms of possible fill oil loss documented by Nuclear Regulatory Commission Bulletin 90-01. The bulletin considers a fill oil loss failure to be a Rosemount design inadequacy.	Replaced with the same model Rosemount transmitter from a different manufacturing lot. The original flow transmitter was sent back to Rosemount to be tested for possible design or manufacturing flaws.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RFW-DPT-4A	Feedwater	With the plant in normal full power operation, operators observed the reactor feedwater level indicator "606A" would occasionally drift as low as 23" reactor level (normal is 35"). This resulted in a power spike due to level fluctuations since the level indication has inputs to the feedwater pump control and is, therefore, susceptible to subsequent reactor level changes. Operators tagged the "A" channel out of service.	A flow transmitter downstream of the level indicator was found to be exhibiting signs of a possible fill oil loss failure as documented in the Nuclear Regulatory Commission Bulletin 90-01.	The transmitter was replaced with a new same type Rosemount transmitter from a manufacturing lot subsequent to the fill oil loss problem. The original transmitter has been sent to Rosemount for testing and evaluation.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RFW-DPT-803C	Feedwater	Twenty four month calibration performed on the "C" main steam line flow transmitter showed the instrument would not calibrate within acceptance tolerances. The system was in a maintenance mode during the annual refueling outage. As found values were out of specification below the manufacturer's recommendations. No Technical Specification requirements are associated with this instrument.	Amplifier circuit board and calibration circuit board, which are piece parts of the transmitter, were found defective. Most likely due to wear-out from age.	Removed defective circuit boards and replaced with new same type boards. Performed recalibration and returned instrument to service.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN- TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RRC-PT-14B	Reactor Protection	This transmitter, which serves as the flow biased average power range monitor trip signal to the reactor protection system, was identified by Rosemount as coming from a manufacturing lot identified with a high failure rate due to loss of fill oil as addressed in Nuclear Regulatory Commission Bulletin 90-01. This transmitter was replaced as a preventive measure.	This is a suspected fill oil loss failure as addressed in Nuclear Regulatory Commission Bulletin 90-01.	Transmitter was replaced with a new same type transmitter. The original transmitter was removed and sent to Rosemount for testing to confirm a fill oil loss condition.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RRC-PT-14D	Reactor Protection	This transmitter, which provides flow biased average power range monitor trip signal to the reactor protection system, is one identified by Rosemount as coming from a manufacturing lot with a high failure rate due to loss of fill oil as addressed in Nuclear Regulatory Commission Bulletin 90-01. This did not fail, but was removed from service in accordance with Rosemount's recommendation.	This is a suspected fill oil loss failure as addressed in Nuclear Regulatory Commission Bulletin 90-01.	Transmitter was replaced with new same type transmitter. The removed transmitter will be sent to Rosemount for testing to confirm a fill oil loss condition.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RRC-PT-24B	Reactor Protection	This transmitter, which serves as the flow biased average power range monitor trip signal to the reactor protection system, was identified by Rosemount as coming from a manufacturing lot identified with a high failure rate due to loss of fill oil as addressed in Nuclear Regulatory Commission Bulletin 90-01. This did not fail, but was removed from service in accordance with Rosemount's recommendation.	This is a suspected fill oil loss failure as addressed in Nuclear Regulatory Commission Bulletin 90-01.	Transmitter was replaced with new same type Rosemount transmitter. The removed transmitter will be sent to Rosemount for testing to confirm a fill oil loss condition.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-LT-26C	Nuclear Steam Supply Shutoff (NSSS)	While conducting the yearly channel calibration performed during the refueling outage on the reactor pressure vessel level transmitter loop "C", technicians noted the transmitter was exhibiting a sustained drift. This is one of the symptoms of fill oil loss a Rosemount transmitter may exhibit immediately prior to failure.	Fill oil loss in Rosemount transmitters are caused by a manufacturing defect as outlined in NRC Bulletin 90-01.	Installed new transmitter and tested per surveillance procedure. This is a suspected fill oil loss failure. The original transmitter was sent to Rosemount for testing.
RCIC-P-3	Reactor Core Isolation Cooling	Plant in Mode 1, 87, 5% power "Coastdown" to scheduled refueling outage. Control room received a "Reactor Core Isolation Cooling Water Leg Pump Motor Overload" alarm and pump trip. Operators closed the trip throttle valve and declared system inoperable.	Bearing failed due to improper adjustment on the bearing oiler such that essentially no oil was available for lubrication. This caused subsequent pump shaft damage.	Installed new pump shaft, inboard and outboard bearing and seal covers. Returned pump to service and performed a visual leakage test with no leakage observed. All replacement parts were the same type as the original.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
DO-P-1A	Diesel Fuel Oil	With the plant in Mode 5, refueling outage, operators placed the diesel oil fuel transfer pump into service for sampling of the diesel day tank fuel. While the pump was running it exhibited excessive noise and vibration. The pump was subsequently taken out of service for repair.	The impeller was found rubbing on the pump bowl because the thrust retaining bolt had become loose when the locking tab had come off as a result of incorrect positioning of the coupling spider. Most likely due to previous maintenance.	The pump was reassembled with new gasket material and the adjusting nut lock tab and coupling spider were reinstalled correctly. System effect was degraded channel since the pump would have been able to operate, but not at its designed performance capacity.
SLC-P-1B	Standby Liquid Control	During normal full power operation, personnel on management overview tour noted an oil leak from the standby liquid control "B" pump head area.	Crankcase cover bolts were stripped and allowed leak through. Cause of stripped bolts most likely due to wear-out from use.	Disassembled crankcase cover and cleaned gasket seating surfaces. Removed worn gasket and replaced with same type new gasket. Replaced stripped crankcase bolts with new longer bolts and torqued to 90 inch-pounds. Verified no visible leakage.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-RV-1B	Main Steam	During annual channel calibration surveillance performed during the refueling outage, the main steam relief valve "1B" would not open by means of its handswitch located in the main control room. This resulted in a loss of the "1B" relief valve, but there was no significant effect on the plant due to system design redundancy.	Plant electricians found a pin in the electrical "Cannon" plug connector pushed back such that no continuity to the solenoid was present. Cause of misplaced pin was unknown.	The connector was disassembled and the pin pushed back in place. The connector was reinstalled and the surveillance was completed satisfactorily.
RCIC-V-111	Reactor Core Isolation Cooling	With the reactor at 8 percent power and holding, awaiting the reactor core isolation system operability test performance following repair on an isolation valve, the 2" check valves on the turbine exhaust vacuum relief line were found stuck closed. Valves are required open for increased power operation, so plant startup was delayed for valve repair.	Valve internals were stuck due to lack of adequate lubrication and dirt buildup, cause unknown.	Valve was disassembled and internals were cleaned and lubricated. Valve was reassembled and flow verified.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RCIC-V-112	Reactor Core Isolation Cooling	With the reactor at 8 percent power and holding, awaiting the reactor core isolation system operability test performance following repair on an isolation valve, the 2" check valves on the turbine exhaust vacuum relief line were found stuck closed. Valves are required open for increased power operation, so plant startup was delayed for valve repair.	Valve internals were stuck due to lack of adequate lubrication and dirt buildup, cause unknown.	Valve was disassembled and internals were cleaned and lubricated. Valve was reassembled and flow verified.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RCIC-V-8	Reactor Core Isolation Cooling	Post modification testing on the reactor core isolation cooling steam supply to the turbine isolation valve required the performance of the Technical Specification surveillance test on the valve resulting in a 10.8 second closing time. The procedural action range, based on ASME pump and valve in-service test program criteria, is less than or at 10 seconds. During review and approval of the surveillance procedure, no one noticed the excessive closing time which should have resulted in declaring the valve and the system inoperable with a 14 day limiting condition for operation, and a 4-hour valve limiting condition for operation.	A routine review of the procedure performed on August 18, 1990 discovered the excessive closing time data. After evaluation and retesting the valve was declared inoperable. Had the valve been required to perform its safety function, this condition alone would not have resulted in a failure of the valve to isolate.	Troubleshooting efforts identified a lack of sufficient lubrication on the valve stem from inadequate maintenance. Valve stem was lubricated and the operability surveillance test was performed satisfactorily.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RHR-V-111A	Residual Heat Removal/Low Press Injection	During scheduled refueling outage, with the "A" loop residual heat removal system in maintenance/test mode, operators on tour noted an excessive packing leak from the 14" gate valve on the return to the reactor pressure vessel. Operators noted the valve probably needed packing since the follower had bottomed out.	Packing material had worn out.	Removed a couple of layers of the old packing and repacked with new material in a sufficient amount to return the packing flange to its correct position. Reinstalled packing flange and torqued gland nuts. Performed a visual leak test with no evident leakage.
DSA-RV-9B	Diesel Starting Air	During performance of annual lift test, the relief valve servicing the diesel starting air tank "5B" exhibited leakage past the seat.	Valve seating surfaces were fouled with rust/corrosion and would not seat properly.	Valve was disassembled, cleaned, and inspected in accordance withplant procedures. Valve seat was lapped and valve was reassembled and retested satisfactorily.
DSA-V-32A	Diesel Starting Air	With the plant in normal full power operation, operator on tour noted the drain valve for the diesel starting air receiver would not close when manually operated.	Exact cause of valve sticking open was unknown.	Faulty valve was removed and replaced with new same type valve. System was pressurized and visually checked for leakage.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
SLC-RV-28B	Standby Liquid Control	During annual safety relief valve testing performed during the scheduled refueling outage, the relief valve for standby liquid control pump "1B" showed leakage past the seat.	Seat leakage was due to nozzle damage resulting from foreign material between the nozzle and disc.	Seating surfaces were machined, valve was bench-tested satisfactorily, and reinstalled in the system.
SLC-RV-29B	Standby Liquid Control	During annual safety relief valve testing performed during the scheduled refueling outage, the relief valve for standby liquid control pump "1B" showed leakage past the seat.	Seat leakage was due to nozzle damage resulting from foreign material between the nozzle and disc.	Seating surfaces were machined, valve was bench-tested satisfactorily, and reinstalled in the system.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
CAC-FCV-1A	Combustible Gas Control	During local leak rate testing performed during the annual refueling outage, the containment atmospheric control drywell suction outboard isolation valve did not meet local leak rate acceptance criteria. Leakage criteria for the entire containment isolation system was within specification, so there was no system or plant effect.	Valve seating surfaces were found not mating properly, most likely due to wear-out.	Valve was disassembled and the seating surfaces were lapped. A blue check was performed to verify a good seal. Valve was returned to service and a satisfactory local leak rate test was performed.
CAC-FCV-1B	Combustible Gas Control	During annual local leak rate test of the containment atmospheric control containment isolation valve, the valve exhibited high-leakage. Although this valve leakage was out of specification for leakage, the total containment leakage was within specification so there was no system or plant effect.	Valve seating surfaces were not mating properly to provide a good seal, most likely due to wear-out.	Valve was disassembled and seating surfaces were lapped. A blue check was performed to assure a good seal. Valve was returned to service and a satisfactory local leak rate test was performed.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
CAC-PCV-4B	Combustible Gas Control	With plant in scheduled refueling outage, the containment atmospheric control discharge to the wetwell outboard containment isolation valve showed leakage through the seat during performance of the 24 month leak test surveillance.	Valve seating surfaces were not mating properly due to wear-out caused by corrosion.	Valve was disassembled and seating surfaces were lapped. Valve was reassembled and leak test was performed satisfactorily.
CAC-V-13	Combustible Gas Control	With the plant in an annual refueling outage, the motor operator for the containment atmospheric control 4" gate valve to containment tripped its overloads with the valve 10 percent from closed seat indication. This resulted in a loss of one of the four trains serving containment, but no significant plant effect. The system was in maintenance at the time of discovery.	Inspection of the valve revealed a small piece of brass imbedded in the root thread of the valve stem causing the stem nut to gall the stem. Cause of brass imbeddment was unknown.	The brass imbeddment was removed and valve stem threads were cleaned. The valve was electronically stroked and stroke time verified within Technical Specification acceptance range.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
HPCS-V-23	High Pressure Core Spray	During performance of the high pressure core spray system operability surveillance test, the test return valve to the suppression pool failed to go full shut. The valve indicated full closed, but the minimum flow valve did not come open and flow indication did not go to zero. This caused the diversion of system flow from the in-vessel spray.	The root cause of this event was an inadequate torque switch setting on the valve.	Faulty valve was temporarily isolated to prevent system flow diversion. The torque switch setting was increased and the valve was satisfactorily tested.
MS-MO-160C	Main Steam	During normal full power operation, operator on turbine building tour observed oil leakage at the main steam bypass valve hydraulic operator. Oil was leaking from the fitting going into the top of the accumulator.	O-rings had evidence of wear due to cyclic aging.	Disassembled fittings at accumulator and replaced o-ring with same type. Reassembled and verified no leakage visible.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MS-MO-68	Main Steam	During the annual refueling outage, operators attempted to open the main steam line outboard drain valve to drain to the condenser from a handswitch in the main control room. The valve blew all three main line fuses as shown by loss of indication in the main control room. This resulted in a loss of the remote operation of the outboard drain function, but no significant plant effect.	Electricians found the local position indicator at the valve operator was showing reverse indication. The indicator consists of 2 nuts with a washer in between, the nuts were loose; most likely due to earlier work performed on the operator this same outage.	The indicator was positioned to show correct position indication and the nuts were tightened. The valve was stroked electrically to verify proper operation.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
MSLC-MO-1A	Main Steam	During performance of annual motor inspection, testing, and maintenance procedure, a short to ground was found while meggering the motor for the main steam leakage control valve "1A" (inboard exhaust to the reactor building). System design is such that system remains functional with loss of a single component, so this failure resulted in degraded performance for the "A" train.	Unscheduled splice was found with pinched wires due to incorrect installation.	Damaged section of cable and splice were removed and a new splice was performed. The procedure was then completed satisfactorily.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
RCIC-MO-8	Reactor Core Isolation Cooling	During performance of a quarterly surveillance performed at full power, it was discovered that reactor core isolation cooling inlet turbine valve closing time was less than the value listed in the Technical Specifications. Although the reaction closing time was slower than required, it did not affect system or plant operation. The valve demonstrated full opening and closing capability.	The cause was unknown, but suspect limit switch was out of adjustment.	Limit switch number 8 of rotor 2 was adjusted to 94 percent of full stroke to pass the surveillance requirements. An engineering analysis was performed to change the valve stroke length to provide both satisfactory stroke times and adequate steam supply.
RHR-MO-16A	Containment Spray	Plant operating at 100% power. During performance of quarterly surveillance on loop "A" residual heat removal system operability, the motor operator for the upper drywell spray outboard isolation valve would not close completely electrically. Operators manually closed the valve and de-energized the valve operator.	Plant electricians disassembled the valve operator and found torque switch contact fingers out of alignment, preventing contact closure. Cause of alignment problem unknown.	Removed and inspected torque switch for any other damage. Replaced the same torque switch in the operator, installing it with proper alignment. Verified operator opened and closed the valve.

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAIN-TENANCE	SYSTEM	PROBLEM DESCRIPTION	CAUSE	ACTION TAKEN
HPCS-MO-11	High Pressure Core Spray	With the plant in a scheduled refueling outage, operators were performing the suction transfer operability surveillance when the high pressure core spray return to the condensate storage tanks valve operator kept tripping the breaker when the valve went in the open direction.	Valve was found stopping approximately six hand wheel turns from backseat. Cause of the valve being out of adjustment is unknown. This does not reduce the flow to the core spray in that there is another isolation valve upstream of this valve that was functional.	Limit switch was adjusted to stop open movement at 90.5 percent. Valve was satisfactorily tested.

2.5 INDICATIONS OF FAILED FUEL

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

INTRODUCTION

A visual inspection of discharged fuel from WNP-2, Cycle 5, was performed from July 31 to August 3, 1990. The purpose of the inspection was to verify assembly and fuel rod structural integrity of discharged fuel. In addition, a visual inspection of selected discharged fuel channels was performed at the same time. An inspection of the suspected Advanced Nuclear Fuels (ANF) leaker was performed by ANF on May 22-24, 1990. Inspection of the three suspected General Electric (GE) leakers was performed by GE on July 26-30, 1990. The results of these inspections are summarized herein.

SUMMARY OF INSPECTION RESULTS

Inspection of suspected leaker XN-1114 by ANF identified fuel rod A07 as failed and A08 as damaged. The appearance and position of the damaged areas indicate fretting from a non-fuel-related object.

Inspection of suspected leaker LJT799 by GE identified rod F02 as failed. The failure mechanism appeared to be accelerated corrosion or Crud Induced Localized Corrosion (CILC). The remainder of the assembly appeared normal.

Inspection of suspected leaker LJT653 by GE identified rod C1 as failed. The C1 rod was broken between seven and ten inches above the lower end plug. The specific failure mechanism has not been identified.

Inspection of suspected leaker LJT749 by GE identified rod D6 as failed. The cladding was heavily corroded and spalled. The failure mechanism appears to be (CILC).

A total of eight assemblies and three channels discharged at the end of Cycle 5 were inspected. No evidence of mechanical damage, geometric distortion or rod bow was observed. All rods inspected appeared properly seated in the lower tie plate. All spacers appeared to be in proper position. The fuel cladding was covered with heavy crud and, in some instances, showed signs of slight spalling.

Inspection of the three channels revealed heavy crud deposition and some evidence of corrosive interaction with the upper guide. No evidence of unusual behavior was observed.

2.6 10CFR50.59 CHANGES, TESTS, AND EXPERIMENTS

Federal Regulations (10CFR50.59) and the Facility Operating License (NPF-21) allow changes to be made to the facility and procedures as described in the Safety Analysis Report and tests or experiments to be conducted which are not described in the Safety Analysis Report without prior Nuclear Regulatory Commission (NRC) approval, unless the proposed change, test or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question. In accordance with 10CFR50.59, summaries of the permanent design changes and temporary plant modifications completed in 1990 are provided. Included are summaries of the safety evaluations.

2.6.1 PLANT MODIFICATIONS

Permanent Plant Modifications at WNP-2 are implemented with a Plant Modification Request (PMR). The following PMRs implemented in 1990 required a Safety Evaluation in accordance with 10CFR50.59. Each permanent change was evaluated and determined not to represent an Unreviewed Safety Question nor require a change to the WNP-2 Technical Specifications.

2.6.1.1
PMR 85-0728
SCN 90-23
PPM 2.7.1

This PMR modified the Control Room HVAC System. Control Room Humidifiers were disabled and the Control Room HVAC Procedure (PPM 2.7.1) was revised due to operational problems with the units. Excessive condensation in the supply ducting and subsequent "rain-out" of the supply register would occur during Humidifier operation.

The FSAR states that the Control Room humidity would be controlled by Humidifier WMA-HU-55A/B between 30 and 50% during normal operation. A Hygrothermograph was used to collect humidity data in the Control Room. The ventilation system is always in the cooling mode; therefore, the humidity is naturally stable. The data collected indicated 17 to 30% relative humidity. The FSAR limits are for personnel comfort. The American Society of Heating, Refrigerating, and Air-Conditioning Engineers (ASHRAE), which sets the HVAC standards, recommends 20 to 60% Relative Humidity for human comfort.

Safety Evaluation Summary

Disabling the humidifiers is a modification that did not result in a change to the WNP-2 Technical Specifications and the Unreviewed Safety Question evaluation concluded that (1) the performance of the Control Room ventilation 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 evaluation were not changed.

2.6.1.2
PMR 85-0744

Plant Modification 85-0744 was initiated as a product improvement modification to the HPCS Diesel Air Start system's pressure switches for the air compressors.

The set points for Diesel Start Air Pressure Switches 15 and 16 (DSA-PS-15 and 16) were changed in a more conservative direction as a result of the deadbands of the new switches. This prevented excessive wear on the diesel driven air compressor by allowing the motor driven air compressor to maintain more air in the tanks. It also coordinated the alarm switches better to the compressor switches.

Safety Evaluation Summary

The Safety Evaluation showed that the proposed change would not increase the probability or consequence of an accident or malfunction of equipment important to safety since the change increased the overall reliability of the HPCS Diesel Air Start System.

2.6.1.3
PMR 85-0743-3

Main Steam Valve 22A (MS-V-22A) stuck in the close position in 1988. This PMR was the modification to the internals of the remaining four Main Steam Isolation Valves (MSIV) MS-V-22B & C and 28 B & C. The first four valves were modified during the 1989 refueling outage. The modification made the valve operation smoother and more reliable.

The change decreased the amount of friction that must be overcome to close the valve and reduced the chances of galling of the valve body bore. The new disk-piston assembly and stem/stem-disk assembly are lighter than the original valve internals. The new disk piston also has a grey cast iron rider ring.

Safety Evaluation Summary

The replaced valve internals improved the overall reliability of the valves and did not affect the valve function. No change to the WNP-2 Technical Specifications was necessary since the Technical Specifications do not describe the internal details of the assembly. There was no Unreviewed Safety Question since the function of the valve did not change and probability of failure decreased.

2.6.1.4
PMR 89-0198

Plant Modification 89-0198 was initiated to modify the cooling water supply to the Division I and II Emergency Diesel Generators. Normally closed air operated butterfly valves SW-V-214, 215 (Div I) and SW-V-216, 217 (Div II) were completely removed from the cooling water supply lines to their respective diesel jacket water heat exchangers to enhance diesel reliability. ASME qualified spacers (i.e. "spool pieces") were installed in place of the wafer style butterfly valves.

One of these four identical valves previously had suffered a disc-to-stem separation because of corrosion of the pins that held the disc to the stem. In evaluating alternative design changes to correct the problem, it was determined that the valves actually served no purpose relative to either the diesel generator or the cooling water system (the diesel keep-warm system is unaffected by the presence of cooling water through the service water side of the jacket water heat exchanger). The valves had been supplied as a part of the diesel skid because the diesel vendor anticipated that, depending upon the design and limitations of the customer's cooling water system, it might be desirable to isolate the cooling water from the diesel when the diesel was not running.

Safety Evaluation Summary

Because the WNP-2 emergency service water system is dedicated to safety-related loads and is properly balanced when all loads are in service, isolating the water when the diesel was not running served no real purpose. Additionally, from a diesel reliability standpoint, having the valves closed actually posed an

unnecessary challenge to diesel's ability to meet its safety function since the valves had to open from their normally-closed positions for the diesel to receive adequate cooling. Deleting the valves solved the corrosion problem with the added benefit of enhanced diesel reliability. This modification did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because the margin to safety as defined in the basis of any Technical Specification was not reduced, nor was the possibility of a different accident or malfunction as previously evaluated in the FSAR created. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR was also not increased.

2.6.1.5
PMR 86-501

This Plant Modification removed all controls, displays, and alarms associated with the Steam Condensing mode of Residual Heat Removal (RHR) operation. The Steam Condensing mode of RHR operation will not be used at WNP-2 and these deactivated controls, indications and alarms in the control room and remote shutdown panels were no longer required.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because: (1) the margin of safety in the Technical Specifications was not reduced by the removal of these deactivated components, and (2) the analysis in the FSAR did not take credit for the Steam Condensing mode of RHR operation.

2.6.1.6
PMR 87-284-0

The Emergency Control Room Chillers would auto start when all permissives were met. This resulted in unwanted Emergency Control Room Chiller starts. The change provided a manual switch in the Control Room to remotely start the chillers. This change only affects how and when the Emergency Control Room Chillers are to be started and does not affect the actual operation of the chillers.

Safety Evaluation Summary

This modification did not result in a reduction in the margin of safety to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. The analysis concluded that a manual start of the fans could be performed by Plant Operations with no decrease in overall plant safety.

2.6.1.7
PMR 88-0003

This PMR changed the power source to the "Z" Signal Trip Units. The "Z" signal trip units are radiation monitors that have GM detectors mounted in the Reactor Building Ventilation exhaust plenum. They provide a trip signal to relay logic in Relay Cabinets RC-1 and RC-2. These Relay Cabinets initiate an NSSSS Group 3 (Reactor Building and Balance of Plant Containment) Isolation.

This change was made to eliminate ESF actuations that occurred when power was lost to the "Z" signal trip units. Before, the change power was supplied from the Reactor Protection System (RPS) Motor Generator Sets A and B. When an "MG set trip" occurred, a "Z" signal trip would be initiated. This provided unnecessary actuation of equipment such as Standby Gas Treatment, shutdown of normal Reactor Building Heating and Ventilating Systems, and startup of ECCS Motor Control Center Room Fans. This change provided divisional uninterruptable power from Inverters IN-2 and IN-3; thereby preventing unnecessary actuation. The change also corrected some problems with the implementation of fail-safe design requirements.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. This change did not change any of the system functions. It improved the reliability of the power supplied to the trip units and corrected the discrepancies between the present configuration and the WNP-2 design requirements for fail safe circuits.

2.6.1.8
PMR 88-0038-04

Plant Modification 88-0038-04 replaces existing high maintenance recorders with current technology-based low maintenance units. The following process recorders were replaced:

1. PI-COMP-TR618A, spared in place trend recorder
2. RRC-FR-614, Recirculation Flow.
3. RRC-TR-650, Recirculation Pump Suction Temperature.
4. RCWU-CR-601, Reactor Water Cleanup Inlet Conductivity. Conductivity High/Low alarms functions are now generated internal to the recorder.
5. RWCU-CR-603, Reactor Water Cleanup Outlet Conductivity.

The modification changes are Class II and Seismic 1M that follow the original design intent. The new equipment is mounted with approved seismic mounting and hardware.

Safety Evaluation Summary

This modification has followed current design criteria for seismic mounting, separation, and isolation. Therefore, the addition of the new recorders did not result in a change to the WNP-2 Technical Specifications or result in an Unreviewed Safety Question because the margin of safety was not reduced or the possibility of an different malfunction as defined in the basis for any Technical Specification was not increased.

2.6.1.9
PMR 88-0038-06

Plant Modification 88-0038-06 was initiated to remove AR-RR-21, Mechanical Vacuum Pump recorder, and install a new recorder, SW-RR-1 in the same panel. The new recorder monitors the Mechanical Vacuum Pump outlet (moved from AR-RR-21) and also the liquid radwaste discharge and service water effluent (RHR Loop A).

This modification makes the information required for 10CFR20 monitoring more readily available with the addition of the new recorder.

Safety Evaluation Summary

This modification has followed current design criteria for seismic mounting, separation, and isolation. Therefore, the addition of the new recorder did not result in a change to the WNP-2 Technical Specifications or result in an Unreviewed Safety Question because the margin of safety was not reduced, nor was the possibility of an different malfunction as defined in the basis for any Technical Specification increased.

2.6.1.10
PMR 88-0041

Plant Design Change 88-0041 was initiated to modify two of eight Primary Containment annulus drain lines, removing the existing blind flanges and replacing them with a flange and 3/4" drain valve on each drain line. The other six drain lines were previously modified in 1984 to include drain valves such that, in the closed position, the ECCS pump rooms would remain isolated from each other in case of room flooding, but would allow periodic monitoring of the drain lines (by opening the drain valves) to check for evidence of water in the annulus sand pocket region. The two lines modified under Design Change 88-0041 were not modified in 1984 due to their location. These lines terminate in the Reactor Building crane bay and, as such, would provide a direct path from Secondary Containment to the outside environment if the drain valves (if present) were open and the Reactor Building crane bay doors were open. However, in light of recent concerns over the possibility of Primary Containment wall corrosion in the sand pocket region (NRC Generic Letter 87-05), WNP-2 determined that a means to periodically monitor the drain lines in the crane bay was necessary, and that administrative means would be employed to assure that Secondary Containment would not be breached by the opening of the new drain valves when the crane bay doors are open.

This modification provided a means to monitor the sand pocket region of the south quadrant of the Primary Containment annulus on a frequent basis without removal of a blind flange. Equipment Operator rounds include a weekly verification that no water is present in these two (and the other six) drain lines. The modification consisted of replacing the altering the existing blind flange design, adding a welded pipe nipple and 3/4" globe valve to each location.

Safety Evaluation Summary

This modification did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because the consequences of an accident (affecting offsite dose limits) are not changed. The added drain valves are locked-closed valves and have caution labels mounted above them stating "Do Not Open This Valve When The Crane Bay Doors Are Open". Additionally, the applicable plant procedure governing the opening of the crane bay doors when Secondary Containment is required (Operating Conditions 1, 2, & 3) was revised to include verification of the locked closed status of these drain valves prior to opening the crane bay doors.

2.6.1.11
PMR 88-0264-1

PMR 88-0264-1 was initiated to provide power and control for the motor operator for non-safety related valve RFW-V-14. Position limit switch contacts on Class 1E valves RFW-V-65A and 65B were used to restrict operation of RFW-V-14. The interlocks with valves RFW-V-65A and 65B will eliminate a potential reactor vessel drain down path which could occur during Long Cycle RWCU System operation if valves RFW-V-65A or 65B were opened while RFW-V-14 was opened.

Safety Evaluation Summary

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 nor was the possibility of a different accident or malfunction as defined in the basis for any Technical Specification increased.

2.6.1.12
PMR 89-0021

This Plant Modification changed the Anticipated Transient Without Scram /Recirculation Pump Trip (ATWS/RPT) logic from a one-out-of-two to a one-out-of-two twice actuating device logic. In addition, a contact was added to the test switch circuit to allow testing of the 147 tripping relay without actually tripping the respective recirculation pump.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because the Technical Specifications did not describe the logic to be used and the change decreased the probability of system failure.

2.6.1.13
PMR 88-0299-16

Plant Modification 88-0299-16 was initiated to relocate Main Steam Pressure Transmitters (MS-PT-8A, B, and C), and add piping supports inside the condenser. These pressure transmitters are non-safety related and measure main condenser absolute pressure. The change did not degrade or affect any Class I system. The Unreviewed Safety Question Analysis was required by procedure because FSAR figure 3.2-23A, Turbine Main Exhaust and Steam System, was changed to show the new pressure tap locations for the transmitters. The original design intent was maintained.

Safety Evaluation Summary

This modification will not result in a change to the WNP-2 Technical Specifications or result in an unreviewed safety question. This modification maintains the original intent to monitor condenser pressure. The addition of supports inside the condenser will lessen the probability of an already analyzed condition.

2.6.1.14
PMR 90-0026

The existing deactivated CIA compressors on 501' elevation of the reactor building were occupying valuable space. It was determined that the area was required for better access to the drywell during outages and this Plant Modification Record (PMR) was implemented to remove the compressors.

Safety Evaluation Summary

The CIA compressors were designed to be the backup system for the cryogenic nitrogen source. During the 1988 refueling outage the CIA compressors were eliminated as a supply source by capping the common discharge line. The compressors are not safety-related and are not addressed in the Technical Specificatiions. The physical removal of the compressors does not increase the probability of an accident or create a different type of accident.

2.6.1.15
PMR 90-061
PER 90-109

During reanalysis of the RFW System piping for snubber optimization inside primary containment, it was discovered that the Architect/Engineer's (A/E's) analysis for determination of operational clearances between the RFW pipe whip restraints and RFW piping had neglected to consider certain thermal transient events which occur periodically during normal plant operation. When the piping thermal movements under these transient conditions are factored into the clearance calculations, it is revealed that there is a possibility that RFW piping could contact the pipe whip restraints during some plant shutdowns, startups and scram events. A Justification for Continued Operation (JCO) was completed as part of Problem Evaluation Request (PER) 90-109. This JCO concluded that Plant Operations could continue until the refueling outage in April 1990.

The detailed RFW System analysis indicated that the calculated interferences, between the piping and whip restraints, did not violate any ASME Code allowable piping stress or WNP-2 loading criteria. Increased RPV nozzle loads were also evaluated and accepted by GE.

During the R-5 outage, all RFW pipe whip restraints were examined for interferences. No observable damage or linear indications were found from extensive NDE (i.e. visual, MT, and UT examinations) of the piping at critical welds and PWS structures as well as at the piping attachment to the RPV at two critical nozzles. The thermally interfering PWS structures were removed from the system by eliminating pipe break locations by means of implementation of the newly refined RFW thermal transient load definition. Based on the NDE results and calculations completed by the Supply System and General Electric, it is concluded that no damage has resulted to either the RPV or the RFW piping. During the Spring 1990 Refueling Outage Plant Modification Record (PMR) 90-061 was implemented to remove or disable selected pipe whip restraints to eliminate the pipe clearance problem.

Safety Evaluation Summary

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. The Technical Specifications do not address margins of safety relative to pipe break postulation. This modification reduces the possibility of a pipe break by significantly increasing the RFW system's structural integrity and; therefore, the overall plant safety margin relative to LOCA postulation is increased.

2.6.1.16
PMR 90-0171

This Plant Modification Record (PMR) was implemented to assure the overload protection would not interfere with the safety function of the Containment Recirculation Fans (CRA-FN-3A, 5A, 5B, and 5C) under accident loading with maximum or minimum voltage at the motor. This change provided a heater one size larger than described in the FSAR for the thermal overload protection relays.

Safety Evaluation Summary

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 accident or malfunction as defined in the basis for any Technical Specification was not increased.

2.6.1.17
PMR 89-034
SCN 90-078

This change provides the utility interface (electrical, plumbing, fire protection, etc.) for the Plant Engineering Center at WNP-2 which will provide permanent housing for a large fraction of the Technical Personnel supporting Plant Operation. It also changes some of the descriptive material provided in Section 2.1.1.1 of the FSAR.

Safety Evaluation Summary

The safety evaluation reviewed the utility interface connections between the new Plant Engineering Center and WNP-2. All changes to Plant Design were found to be Quality Class II and the interfaces had no effect on any safety-related plant systems.

2.6.2 LIFTED LEADS AND JUMPERS (Temporary Changes)

The following are summaries of temporary 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.

2.6.2.1

LLJ 90-014

The purpose of this temporary change was to allow de-energization of Main Steam Line Radiation Monitor (MS-RIS-601B) without tripping the B Gland Seal Exhaust Fan. The change was accomplished by jumpering out the contacts of relays K70 and K73 for the gland exhaust fan trips on the Main Steam Rad Monitor Channel B.

Safety Evaluation Summary

This change did not require either a Technical Specification change or result in an Unreviewed Safety Question because the Technical Specifications do not discuss this particular trip, credit is not taken for this trip in the Accident Analysis and, if a Main Steam Line High Radiation Monitor trip signal were to be received while these jumpers were installed, the main steam isolation valves would close, isolating the reactor from the condenser. If, because of a single failure, the Main Steam Isolation Valves were to fail to close on the High Radiation signal, the Air Ejector Suction Valve from the Condenser (AR-V-1) would close, also isolating the condenser from the gland exhausters.

2.6.2.2
LLJ 90-019, -089, -098, and -099

These temporary changes are associated with the Air Removal (AR) System. Division I gland seal steam exhauster AR-EX-1A was out of service for corrective maintenance with Division II exhauster in service. Technical Specification required surveillance testing of Division II main steam line radiation monitors MS-RIS-610B and MS-RIS-610D normally requires transfer to the Division I exhauster if the Division II unit is on line to preclude tripping. To allow completion of surveillance testing, the trip function from the monitor under test was defeated.

To allow completion of required surveillance testing, the high-high and inop trip functions from the respective monitor to the exhauster was defeated by jumper installation. The jumper was removed upon completion of the surveillance.

Safety Evaluation Summary

The jumper installation was a temporary change to facilitate testing and the redundant Division II monitor trip functions were still operable. No credit is taken in the Technical Specifications for this trip function and failure of this system to isolate is bounded by failure of the Off-Gas system (FSAR 15.7.1), which is less limiting.

2.6.2.3
LLJ 90-051 and Several Changes to Operating and Surveillance Procedures

This temporary change allowed the High Pressure Core Spray (HPCS) Diesel Generator to be maintained in the droop mode of operation until a new isochronous/droop switch could be procured and installed. Two electrical jumpers were installed within the HPCS Diesel Generator isochronous/droop selection circuitry to place the speed governor permanently in the droop mode of operation. The failure of the switch within the HPCS DG selection circuitry required the installation of the jumpers to assure the unit had the capability to be parallelled to the offsite source for testing purposes. The switch was discovered to have intermittent contacts which, when in the droop mode of operation, could result in the diesel generator to revert to the isochronous mode of operation. This caused the unit to become unstable when paralleled to the offsite source (See LER 90-004 for additional details).

Safety Evaluation Summary

Various test data for the HPCS Diesel Generator were reviewed. This test data showed that the unit design had the capability to meet all regulatory requirements when in the droop mode of operation. Acceptable diesel generator starting times and HPCS pump acceleration capabilities were demonstrated in this mode of operation. The only result of operating in the droop mode of operation is that the steady state speed would be slightly less due to the droop characteristics when the engine was loaded. The droop was set during startup testing at 3.15% for load changes from 0 to 100%. This represented a minimum steady state frequency of 58.2 hertz. This was well within the Technical Specification steady state frequency requirement of 57 hertz.

2.6.2.4
LLJ 90-089

This Temporary Change involved the Air Removal (AR) System. The Division I gland seal steam exhauster AR-EX-1A was out of service for corrective maintenance with Division II exhauster in service. Tech Spec required surveillance testing of the Division II main steam line radiation monitors MS-RIS-610B and MS-RIS-610D normally requires transfer to the Division I exhauster if the Division II unit is on line to preclude tripping. To allow completion of surveillance testing, the trip function from the monitor under test was defeated.

To allow completion of required surveillance testing, the high-high and inop trip functions from the respective monitor to the exhauster was defeated by jumper installation in accordance with approved plant procedures. The jumper was removed upon completion of the surveillance.

Safety Evaluation Summary

The jumper installation was a temporary change to facilitate testing and the redundant Division II monitor trip functions were still operable. No credit is taken in the Technical Specifications for this trip function and failure of this system to isolate is bounded by failure of Off-Gas (FSAR 15.7.1) and is less limiting.

2.6.2.5
LLJ 90-092

This temporary change was made to assure one of the Bypass and Inoperable Status Indication (BISI) alarms associated with the Residual Heat Removal (RHR) "B" System remain operational. The RHR Heat Exchanger "B" Inboard and Outboard Vent Valves (RHR-V-73B and RHR-V-74B) had been de-energized and danger tagged for corrective maintenance. This caused the two BISI relays associated with these valves (RHR-RLY-80/V73B and RHR-RLY-80/V74B) to actuate the BISI alarm "RHR B/C MOV NETWORK PWR LOSS/OL". This, in turn, actuated the alarm on Panel 601 "RHR B OUT OF SERVICE". With this alarm in, all other alarms associated with "RHR B OUT OF SERVICE" would be masked. This temporary change removed relays RHR-RLY-80-V73B and RHR-RLY-80-V74B which cleared the alarms described above.

Safety Evaluation Summary

The removal of these relays will enhance safety by allowing other alarm conditions associated with the annunciators to be communicated to the Plant Operators. The BISI indication is the only function of the two removed relays. The power was removed and clearance tags hung, and further warning is not required on the inoperability of these valves. The valves are shut and de-energized as they are not required to open except for venting or filling operations.

2.6.2.6
LLJ 90-102 and 103
PDF 90-185 and 189

This temporary change was written to allow the Bypassed and Inoperable Status Indication (BISI) to function in a normal manner while maintenance was being performed on the Control Room Chiller units (CCH-CR-1A and CCH-CR-1B). The BISI inputs (CCH-CR-1A PWR LOSS and CCH-CR-1B PWR LOSS) were removed to allow other alarm conditions to be communicated to Plant Operators.

Safety Evaluation Summary
This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. The Chiller Units had the power removed and they were tagged out of service. This change allowed other alarms associated with Control Room Heating and Ventilating conditions to function to alert Plant Operators to system problems.

2.6.2.7
LLJ 90-142, 145, 186, 187, 188, 189, 197, 209, 260, 261, 266, 267, 273

During the annual refueling outage a number of temporary changes are required to support electrical power requirements. This need occurs when a piece of electrical equipment is taken out of service for maintenance during the outage and power needs to be supplied to the loads normally fed by this equipment from another source. The electrical Lifted Leads and Jumpers listed above were written to implement this type of temporary change during the 1990 Refueling Outage.

Safety Evaluation Summary

Each of the above temporary changes were evaluated to assure they did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. The evaluation made sure (1) the margin of safety in the Technical Specifications was not reduced, and (2) the boundary conditions of the FSAR evaluations were not changed. Specific factors considered included fusing in the involved circuits, cable size, divisional separation between circuits, and the status of the plant during the time the temporary change would be in effect.

2.6.2.8
LLJ 90-149

This temporary change involved the deactivation of the Residual Heat Removal (RHR) Suppression Pool Cooling Test Return Valve (RHR-V-21) which was de-energized due to a positioning problem. With RHR-V-21 de-energized, the RHR "B" Bypass and Inoperable Status Indication (BISI) alarm was continuously energized which prevented other inputs to the BISI from warning the Plant Operators of an abnormal condition. This temporary change removed the BISI relay RHR-RLY-80/V21 from its socket and cleared the BISI alarm while maintenance was performed on RHR-V-21.

Safety Evaluation Summary

The relay removed provides indication only. The RHR-V-21 valve which feeds this relay is de-energized in the closed position. Thus, the RHR system was otherwise fully operational and no Technical Specification requirements were affected.

2.6.2.9
LLJ 90-191

This temporary change was made to supply temporary seal cooling water to Air Removal Pump Number 1 (AR-P-1). Water from the Demineralized Water (DW) system was supplied by means of hose connections to maintain seal cooling to AR-P-1. The normal cooling source is from the Turbine Service Water (TSW) System.

Safety Evaluation Summary

This change did not result in a change to WNP-2 Technical Specifications or an involve Unresolved Safety Question since those portions of the Demineralized Water and TSW systems do not affect equipment important to safety and neither system is Technical Specification related.

2.6.2.10
LLJ 90-192, 245, 246, 247, 248

These temporary changes were made during a maintenance activity to change out selected safety related HFA relays during the annual R5 refueling outage. The activity is a result of an on going maintenance program aimed at periodic inspection and maintenance of all Plant-installed, GE-type HFA relays. To implement this change a jumper is used to maintain the continuity of the relay system AC neutral while changing selected relays.

Due to the design of the relay logic AC power distribution system, several relays have AC neutrals that are in parallel with each other. The maintenance activity targeted several relays on the same neutral run. Several other relays that use the same neutral were not involved with the activity. This "Daisy Chain" of neutrals would affect non-targeted relays when targeted relays were determed and removed from the plant. Installation of the jumper allowed the non-targeted relays to remain energized by jumpering the neutral around the targeted relays.

Safety Evaluation Summary

This jumper installation did not result in a change to Plant Technical Specifications or present an Unreviewed Safety Question because the jumper was a temporary installation to keep plant systems in their normal lineup during the on-going maintenance activity. The jumper maintained the system logic in a lineup consistent with the description of the system in the FSAR and allowed the down-stream logic to perform in a manner consistent with Plant Technical Specifications. Those relays that were changed out as a result of the maintenance activity were declared inoperable during the maintenance activity and were not returned to service until after the jumper was removed and operability testing was completed.

2.6.2.11
LLJ 90-214

This temporary change was performed during the refueling outage to allow the Source Range Monitor (SRM) and Intermediate Range Monitor (IRM) Surveillance to be performed to verify operation of the "RODOUT BLOCK" Annunciator. An electrical jumper was used to bypass the "B" Channel of the Refuel-Bridge-Over-Core Interlock to clear the "RODOUT BLOCK" caused by a malfunctioning limit switch in the Refuel Platform.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. The "A" Channel remained operable during this operation. The Refuel Bridge was not over the core and it was tagged out of service to prohibit movement over the core.

2.6.2.12
LLJ 90-231

During routine maintenance activity associated with 4160 volt breaker 8-85/1, two of the three installed degraded voltage relays were damaged. They were installed on the door of a cubicle and were impacted by the circuit breaker during removal. One relay was available in spares, and was installed to replace one of the two relays. Additional spare relays were placed on order. A temporary change was made to place the channels for the inoperable relay in the trip condition.

Safety Evaluation Summary

This evaluation found that this configuration did not constitute an Unreviewed Safety Question because the degraded grid condition continued to be monitored by the remaining relays. The net effect was a logic change from two-out-of-three to one-out-of-two. This jumper was installed in accordance with the direction provided in an action statement within the Plant Technical Specifications.

2.6.2.13
LLJ 90-233

During a power outage on 480 Volt Switchgear Unit 11 (SL-11) temporary power was needed for the Clearwell Transfer Pump (FW-P-3A). This pump receives its power from Motor Control Center 1C (E-MC-1C) which is fed by SL-11. An electrical jumper was provided to supply power to cubicle 4D of MC-1C from Motor Control Center 5A (E-MC-5A).

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. Installing this temporary power source allowed this source of water to remain operable. All circuits involved in this temporary change are non-safety related and the wire installed was sized for the load.

2.6.2.14
LLJ 90-250 and 90-251

This temporary change was made using electrical jumpers to the Plant during the refueling outage. A Local Leak Rate Test (LLRT) required the opening of the High Pressure Core Spray (HPCS) Injection Valve (HPCS-V-4) with a High Reactor Water Level (Level 8) condition. The high water level condition would normally require HPCS-V-4 closure.

Safety Evaluation Summary

At this point during the outage the HPCS Pump (HPCS-P-1) fuses were pulled to prevent the pump from running. In addition, the manual isolation valve between HPCS-V-4 and the vessel (HPCS-V-51) was closed. Thus, the movement of HPCS-V-4 to perform the LLRT could not accidentally inject water in the vessel or drain water from the vessel. Therefore, this activity did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question.

2.6.2.15
LLJ 90-255

The purpose of this temporary change was to allow running the Reactor Recirculation pump on 15 Hz while the Reactor Protective System (RPS) was de-energized for work during the refueling outage. The change was accomplished by installing jumpers in the RRC pump trip circuit.

Safety Evaluation Summary

This change did not require either a Technical Specification change or result in an Unreviewed Safety Question because the pump trip due to valve isolation signal is only required, by the Technical Specifications, to be operational in Modes 1, 2 or 3. These jumpers were only installed during Mode 5.

2.6.2.16
LLJ 90-256

This temporary change removed selected fuses to prevent the Reactor Recirculation pump from tripping on an ATWS signal when the Reactor Protective System (RPS) was de-energized for work during the refueling outage. The change was accomplished by removing fuses F-8 in the ATWS trip circuit of RRC pump B.

Safety Evaluation Summary

This change did not require either a Technical Specification change or result in an Unreviewed Safety Question because the ATWS-RPT is only required to be operable in Mode 1. The Reactor was in Mode 5 for the entire time the fuses were removed.

2.6.2.17
LLJ 90-268

This change was a temporary jumper to provide level indication at the remote shutdown panel while a qualified splice was installed.

Safety Evaluation Summary

The reactor was in a analyzed condition, shutdown, depressurized and flooded. The temporary jumper did not increase the probability of previously-evaluated accidents nor increase their consequences. The temporary jumper did not remove the reactor from its analyzed condition and; therefore, did not create the possibility of an accident of a different type than had been previously evaluated, nor does the jumper reduce any Technical Specification margin of safety.

2.6.2.18
LLJ 90-274 and 90-368

Temporary changes 90-274 and 90-368 were installed in the plant to provide power to the "Hydrobroom" used on the 606 foot level of the Reactor Building. The "Hydrobroom" is used to decontaminate the reactor cavity during the refueling outage.

Safety Evaluation Summary

Temporary power was provided from Motor Control Center 7C-B (MC-7C-B) Cubicle 5E. This Motor Control Center is not safety-related and the change did not increase the probability of occurrence of an accident or malfunction of equipment important to safety. The cable, fusing and MC-7C-B were all sufficient to power the "Hydrobroom" at the required 87 amperes of 480 volt AC power.

2.6.2.19
LLJ 90-280, 281, and 307

The changes were temporary jumpers in local racks at WNP-2. The jumpers were inserted to preclude RPS Channel A trips from RPV level (MS-LIS-24A & MS-LIS-24C) and Drywell pressure (C72-N002C) while installing qualified splices in the rack. An isolation of MSL drain valves from a trip of C72-N002C or B22-N061C was also prevented.

Safety Evaluation Summary

The reactor was shutdown, depressurized and flooded. Temporarily removing one channel of RPV level input from RPS did not alter the capability of scramming. Removing one channel of drywell pressure from RPS did not increase the probability of an accident since the drywell was open, nor were the consequences of any previously evaluated accident increased. Installing the jumpers did not create the possibility of an accident of a different type than has been previously evaluated.

2.6.2.20
LLJ 90-286

The purpose of this change was to add a source of demineralized water to the Control Rod Drive (CRD) system during the CRD system outage such that the Hydraulic Control Units (HCUs) would not become air bound.

During the outage of the CRD pumps and a portion of the system, a mechanical jumper was added to the system such that Demineralized Water (DW) at normal DW water pressures could be supplied to the CRD HCUs by means of the CRD system charging water line. This kept them full and prevented air binding of the accumulators.

Safety Evaluation Summary

No change to the Technical Specifications was required because the system was removed from service. The addition of the jumper did not cause operability problems to a system already declared inoperable. The action statement of Technical Specification 3.1.3.5.B had been completed.

2.6.2.21
LLJ 90-290, 293, 299, 301 and 310

All of the temporary changes listed above were as a result of the Instrumentation Wire Splice Inspection Work. The wire splices were repaired during R5 outage and involved Instrument Racks on the 522 foot level of the Reactor Building. The instruments involved with the splices are connected to the Reactor Protection System (RPS). The method chosen to accomplish the repair was to install a temporary jumper while the instrument was out of service. The jumper was installed to replace the switch or relay contacts in the instrument associated with the SCRAM logic while the splice work was being accomplished. The jumper was removed when the splice work was completed. This was done to avoid an unplanned challenge to the SCRAM logic.

Safety Evaluation Summary

These changes did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. All of the instruments jumpered were monitoring plant parameters that were invalid while the plant is in an outage condition. The signals included signals such as high vessel pressure, which cannot occur while the vessel head is off. Therefore, none of these affected any of the USQ evaluation criteria.

2.6.2.22
LLJ 90-308
PDF 90-576
PDF 90-647

During the Spring 1989 refueling outage (R-4) the flex hose (CIA-FLX-1C) which supplies air to the actuator for Main Steam Relief Valve (MS-RV-2D) was found damaged and was removed from service until a new one could be installed. It was intended to replace the flex hose in the Spring 1990 outage (R-5); however, due to procurement problems, the new flex hose was not available for installation prior to the originally scheduled startup from R-5. Since MS-RV-2D was still capable of opening in the safety mode (relief mode was declared out of service), it was necessary to be able to open MS-RV-2D manually at approximately 10% reactor power to perform the RSV Acoustic Monitor test and calibration as required by Technical Specifications. With CIA-FLX-1C removed, the required surveillance of the Acoustic monitor was not possible; and, therefore, LLJ 90-308 was installed to provide a temporary air supply from RHR-AO-41A to MS-RV-2D to be used to open the SRV manually while the reactor was at 10% power. Air Operator RHR-AO-41A had been previously deactivated and, therefore, its air supply was available for use to temporarily operate MS-RV-2D.

The change disconnected the air supply to RHR-AO-41A. This air supply was then connected to a temporary pipe routed from the vicinity of RHR-AO-41A to MS-RV-2D where it was temporarily connected to MS-AO-2D. The CAS Supply to RHR-AO-41A is isolatable by two locked-closed valves outside containment, so that at all times other than the Acoustic Monitor surveillance testing, the air supply to RHR-AO-41A and temporarily MS-RV-2D is isolated to maintain containment integrity.

The delay in startup from R-5, caused by the rebuilding of an emergency generator, allowed additional time for procurement of the new flex hose, which was subsequently installed. The temporary air line from RHR-AO-41A to MS-RV-2D was removed and LLJ 90-308 and its associated procedure changes, PDF 90-576 and PDF 90-647, were canceled in July, 1990.

Safety Evaluation Summary

The change did not require a Technical Specification change and did not result in an Unreviewed Safety Question because the jumper was for temporary actuation of MS-RV-2D only to meet Technical Specification requirements for the associated Acoustic Monitor; the jumper was isolatable from outside containment by two locked-closed valves to maintain containment integrity and prevent introduction of air into the containment during normal operation; the jumper was supported at suitable intervals to ensure it remains in place during a seismic event; a missile analysis was performed to ensure that if the temporary air line did fall it could not damage any component in its zone of influence; there were no affects that the jumper could produce that would increase the probability or consequences of an accident; there are no new events that the jumper could cause; and use of the jumper would maintain the margin of safety by enabling the calibration of the MS-RV-2D Acoustic Monitor.

2.6.2.23
LLJ 90-312

Required relief valve testing on the "B" RHR loop requires the loop to be drained. The draining of the loop causes the loss of the "B" loop for potential use as an alternate shutdown cooling path. This temporary change provided blank flanges to replace relief valves while the relief valves were being tested.

Safety Evaluation Summary

To allow the "B" loop to be available while the relief valves were being tested, ASME blank flanges were staged and ready to be installed under the guidance of an ASME Section XI plan and a Maintenance Work Request (MWR). In the event that the loop was required, the flanges would have been installed and the loop filled and put into service. These valves provide thermal over pressure protection. At full RHR pump shutoff head, these valves are not required for pressure protection. In the event the loop was filled, manual venting and monitoring of system pressure and temperature would provide adequate protection. This was applicable only in Modes 4 and 5. Therefore, this activity did not result in a change to WNP-2 Technical Specifications or involve an Unreviewed Safety Question.

2.6.2.24
LLJ 90-449

This temporary change provided for opening Reactor Outside Air Damper 3B (ROA-AD-3B) for purging the dry well. The change was made to install a nitrogen bottle to ROA-AD-3B to provide opening air pressure.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unresolved Safety Question because (1) containment isolation capability was provided by Containment Supply Purge Valves 1,2,3, and 4 (CSP-V-1,2,3,and 4), (2) failure of the nitrogen supply would have resulted in the damper failing in the closed position, and (3) ROA-AD-3B is not listed in the Technical Specifications as a containment isolation valve.

2.6.2.25
LLJ 90-484
PER 290-0675

This temporary change corrected a problem with the HPCS 125 volt battery. Weekly battery surveillance discovered that all electrolyte had leaked from cell #9.

A JCO was prepared which concluded that the HPCS battery would be capable of meeting the design requirements with one cell removed, i. e., a 57-cell battery. The evaluation included a review of the most recent battery performance test and the design requirements. The immediate disposition of this item was to jumper cell #9 until a replacement cell could be prepared for installation.

Safety Evaluation Summary

The use of the HPCS battery with 57 cells did not result in a change to the WNP-2 Technical Specifications and the Unreviewed Safety Question evaluation concluded (1) the HPCS battery was capable of performing its design function, (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.

2.6.2.26
LLJ 90-495, 90-496, 90-502, 90-527, 90-560 and 90-566

These temporary changes were associated with a low voltage condition detected by surveillance testing on cells 39 and 40 of battery E-B1-1. The safety function of Battery E-B1-1 is to provide 125 Volt DC power to the Division 1 safety-related loads including critical switchgear control power and various other control functions. Battery E-B1-1 is comprised of 58, type GN-13, cells and is manufactured by Exide Corporation. Each cell is required to produce 2.13 volts of DC power. The temporary changes involved electrical connections to remove the two cells from service, charge the cells, and place the cells back into service.

Safety Evaluation Summary

Cells 39 and 40 could only be removed from service if E-B1-1 remained capable of providing the voltage and current required by its connected loads during malfunctions and accident conditions. The latest surveillance test data was evaluated and analysis was performed which showed the battery would be capable of producing the required 105 Volts at "End-of-Discharge Voltage" conditions with the two cells removed. During cell charging, the safety evaluation provided assurance that the battery would remain functional under all conditions, including seismic events. Finally, during re-connection of the cells an evaluation was performed to ensure that the charger was capable of supplying the load for a short period of time when the battery was taken out of service.

2.6.2.27
LLJ 90-524 and 459A

The valve position signal was temporarily removed from the RCIC-V-8 NOT FULLY OPEN SIGNAL so the DIV 1 OUT OF SERVICE annunciator could be activated by another Bypass and Inoperable Status (BISI) alarm until the valve limit switch could be adjusted. The wire was lifted in the control room that removed the valve position signal. The BISI was still available for a remote manual closure of RCIC-V-8, and the position indication lights in the control room were still available.

Safety Evaluation Summary

This did not result in a change to the WNP-2 Technical Specifications and the Unreviewed Safety Question concluded (1) the operation of RCIC-V-8 was not affected, (2) the valve position can be monitored by the position indication lights, (3) the RCIC DIV 1 OUT OF SERVICE ANNUNCIATOR would be available to monitor the other 13 parameters instead of being in an alarming condition, (4) the margin of safety provided in the Technical Specifications was not changed, and (5) the boundary conditions for the FSAR evaluation were not changed.

2.6.2.28
LLJ 90-559

HPCS-V-23 was declared inoperable due to its failure to fully close during surveillance testing. HPCS-V-23 was subsequently manually closed using the handwheel and de-energized at the MCC by opening its control power breaker. The BISI (Bypass and Inoperable Status Indication) annunciator circuit logic associated with HPCS MOV power network is configured such that,if any HPCS MOV power is lost (as is the case when the MOV power circuit breaker is opened), an alarm occurs. With any HPCS system BISI annunciator illuminated, the "HPCS System Out-Of-Service" annunciator will also illuminate. These two illuminated annunciators, understood by Control Room Operators as being associated with the HPCS-V-23 being de-energized, would mask any other MOV power loss elsewhere in the HPCS system; a condition which could place the HPCS system in a inoperable condition.

Safety Evaluation Summary

A Jumper/Lifted Lead request was approved which removed HPCS-RLY-80/V23 from the BISI circuit, extinguishing the "MOV Network Power Loss" and "HPCS System Inoperable" annunciators even though the HPCS-V-23 power circuit breaker was open. This allowed the BISI circuit to continue to monitor all other MOVs and inoperable status inputs and respond accordingly by annunciation. Removal of this relay from the BISI circuit did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because (1) HPCS-V-23 is an HPCS pump test return line isolation valve required to be closed in order for the HPCS system to perform its safety function, (2) HPCS-V-23 was manually closed and de-energized and as such did not require BISI monitoring for loss of power, and (3) removal of HPCS-RLY-80/V23 allowed the BISI circuit to function as designed with the HPCS-V-23 control power circuit breaker open. Therefore, the margin of safety provided in the Technical Specifications was not changed, and the boundary conditions for the FSAR evaluations were not changed.

2.6.2.29
LLJ 90-606

One of two New Fuel Vault criticality monitors failed and was spuriously alarming. The instrument was deactivated by disconnecting the appropriate amphenol connectors.

Safety Evaluation Summary

The Technical Specifications require operable criticality monitors only when fuel exists in the New Fuel Vault. There is no fuel in the vault, with no plan to utilize the vault in the near future. Therefore, this activity did not result in a change to WNP-2 Technical Specifications or involve an Unreviewed Safety Question.

2.6.2.30
LLJ 90-608

The purpose of this temporary change was to allow operation of the refueling bridge while the over-the-core position switches were malfunctioning. The change was accomplished by installing jumpers on the refueling bridge bypassing the bridge over-the-core position switches.

Safety Evaluation Summary

This change did not require either a Technical Specification change or result in an Unreviewed Safety Question because the over-the-core interlock is designed to prevent withdrawing or installing fuel into the core when a control rod was withdrawn by preventing bridge movement over the core. Moving fuel in the core can only be physically accomplished in Mode 5. The jumpers were only installed in Mode 1.

2.6.2.31
LLJ 90-629

A temporary jumper was installed to allow the installation of a temporary control air dryer to take the place of the nonfunctional permanent dryer. The CAS dryer outlet filters had failed due to overheating. The dryer was taken out of service until the dryer was repaired. Since the cause of the high temperature could not be immediately determined, a new heatless-drying tower was purchased. This temporary tower was connected with hoses to the pre-filters and after-filters.

Safety Evaluation Summary

The use of the temporary dryer did not result in a change to the WNP-2 Technical Specifications and the Unreviewed Safety Question concluded (1) the performance of the Control Air 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.

2.6.2.32
LLJ 90-634

The purpose of this temporary change was to defeat a rod block caused by a defective refueling bridge over-the-core interlock switches. The change was accomplished by installing jumpers in the refueling bridge rod block circuit bypassing the bridge over-the-core position switches.

Safety Evaluation Summary

This change did not require either a Technical Specification change or result in an Unreviewed Safety Question because the over-the-core interlock is designed to prevent withdrawing or inserting a control rod when the bridge is over the core (potentially installing fuel into or removing fuel from the core). Fuel can only moved in the core in Mode 5. The jumpers were only installed in Modes 1 and 2.

2.6.2.33
LLJ 90-651
PER 290-0972

PER 290-0972 was written when a cross-connect was found between the Sanitary Drain (SD) and Reactor Exhaust Systems (REA). This resulted in an unanalyzed bypass of Secondary Containment as reported in LER 90-032.

A Jumper and Lifted Lead (LLJ 90-651) was issued to install a plug in the sanitary drain system at the 439" elevation in the reactor building to prevent a possible unmonitored release of liquid radioactive material after an accident. A PMR was written to permanently eliminate the REA and SD system cross-connect.

Safety Evaluation Summary

This modification to add the temporary plug in the Sanitary Drain System did not result in a change to the WNP-2 Technical Specifications or involve an Unresolved Safety Question because the margin of safety was not reduced and the possibility of a different malfunction as defined in the basis for any Technical Specification was not increased. Secondary Containment integrity has always been maintained and test results have been acceptable. During a Post-Accident Condition the Reactor Building is maintained at a negative pressure.

2.6.3 FSAR EVALUATIONS

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

2.6.3.1
MINOR CYCLE 6 RELOAD CHANGES

The purpose of this change was to accurately describe the reactor core as loaded for Cycle 6. Certain minor changes were required since the Cycle Submittal in the spring.

The changes required and reasons for each are as follows:

A. Two lead fuel assemblies, XN-1163 and XN-1164, contained two segmented rods in each assembly which were not growing as rapidly as the other rods in the assembly. If left in the assembly, these rods had a potential to become loose in the upper tieplate due to improper seating of the endcap in the tieplate. We had planned to replace these rods with inert rods, but the NRC would not accept the vendor's approach to the replacement analysis. However, the NRC would accept the replacement of the segmented rods with natural uranium rods. This replacement was performed during the refueling outage.

B. During the disassembly and reassembly of fuel bundle XN-1163 during the refueling outage, a single finger spring tab became bent over during the underwater channeling process. The only possible repair was to break the tab free of the assembly. Accounting for the bundle's core loading position, Advanced Nuclear Fuels (ANF) performed an analysis which demonstrated that the small increase in bypass leakage through the modified finger spring would not have significant impact on the bundle's thermal hydraulic performance.

C. During the fuel sipping effort performed in the outage, it was determined that bundle XN-1114 had a fuel pin leak. A data search revealed that XN-1029, a bundle planned for discharge, possessed very nearly the same nuclear characteristics as XN-1114. ANF preformed an analysis which showed that the substitution of XN-1029 for XN-1114 in the core loading scheme would have no significant steady state or transient impact on the core.

Safety Evaluation Summary

No change to the Technical Specifications was required and no Unreviewed Safety Question resulted from these changes because the analyses, in all cases, demonstrated that core transient and steady state limits would not be significantly affected.

2.6.3.2
SCN 90-081

This Safety Analysis Report (SAR) Change Notice (SCN) revised the Emergency Preparedness Plan for WNP-2. Numerous updates were made to the Plan to accurately describe the current methods of handling emergencies at WNP-2. This included an update of the titles of State Agencies, removal of the requirement for an Interplant Operations Communications Channel between WNP-1 and WNP-2 since WNP-1 is not operational, movement of the First Aid Facility to a different location, a clarification of methods used for Dose Projection, and numerous other minor changes.

Safety Evaluation Summary

The Safety Evaluation concluded that the changes made enhanced the ability of WNP-2 to respond to emergencies. This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question.

2.6.3.3
SCN 89-063

This SCN revised the FSAR by removing the requirement to perform Local Leak Rate Testing (LLRT) on test, vent and drain connection valves which are located within the Primary Containment boundary.

This change to the FSAR eliminated local leak rate testing on approximately 173 test, vent, or drain valves, all being 3/4" diameter globe valves which are normally closed and capped during power operations. This change was made to reduce outage LLRT efforts and decrease personnel exposure.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications in that these valves were not included in the applicable Technical Specification section which lists the main line containment isolation valves requiring LLRT. This change did not involve an Unreviewed Safety Question because the consequences of an accident was not increased, nor was the margin of safety defined in the basis for any Technical Specification reduced. These test, vent, and drain connection valves are normally closed and capped during power operations and are verified closed every 31 days as required by plant Technical Specifications. The consequences of a accident are not increased by this change in that these valves are tested for leakage as part of the overall Type A Integrated Leak Rate Test and; therefore, do not represent an untested containment isolation boundary.

2.6.3.4
SCN 89-062

This item changed the WNP-2 Physical Security Plan by deleting the requirement that all Supply System employees have a physical examination in order to obtain unescorted access to the Plant.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question. No reasonable relationship existed between the need for a physical examination and unescorted access for non-security personnel.

2.6.3.5
SCN 90-003

This change to the WNP-2 Physical Security Plan deletes the requirement for a separate vital area for the Remote Shutdown Room and allows the securing of Alternate Access (AAP) Point badges and keycards in the Access Control Station when the AAP is not manned.

Safety Evaluation Summary

The Remote Shutdown Room does not need to be controlled as a separate vital area since it is located in a larger vital island. Securing badges and key cards in the AAP Access Control Station when the AAP is unattended enhances the security of badges and keycards located there.

2.6.3.6
SCN 90-084

This SCN revised the content of SAR Chapter 13, Conduct of Operations. It modified this chapter to reflect the current organization and made several other updates to reflect the current Supply System conduct of operations.

Safety Evaluation Summary

Changes were reviewed to assure there was no change to the Technical Specifications and that they did not involve an Unreviewed Safety Question. The independence of the Quality Assurance-related organizations to perform their duties was maintained in the reorganization.

2.6.3.7
SCN 90-086

The Off-Site Dose Calculation Manual (ODCM) was modified to update details of the Radiological Environmental Monitoring Program described in Section 5.0. Tables were updated to reflect the current sampling locations used to support WNP-2 Operations.

Safety Evaluation Summary

The change provided updated maps and tabular data on sample locations. The consequences of an accident are not changed as a result of this update. In addition, the margin of safety as defined in the bases of the Technical Specifications is not reduced.

2.6.4 PROBLEM EVALUATIONS

The Plant Problems-Plant Problem Reports Procedure (PPM 1.3.15) provides instructions for the disposition and documentation of plant problems. 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 did not involve a change to the Technical Specifications or involve an Unreviewed Safety Question.

2.6.4.1
PER 90-017

This Problem Evaluation Report documented an error found in the meteorology calculations in Amendment 36 of the FSAR. The data was found to be non-conservative by approximately a factor of 10. Specifically, the values for the source term dispersion pattern relative concentration factor, X/Q, were specified incorrectly.

Safety Evaluation Summary

A previous Justification for Continued Operation (JCO), Nonconformance Report 288-0357, was revised to address this issue and the offsite consequences were found to be acceptable. The previous JCO reviewed the issue of Secondary Containment draw-down time following a postulated Design Basis Accident (DBA) condition. During this review the incorrect X/Q values were discovered. Further review showed these erroneous X/Q values were not used in support of the Chapter 15 accident analysis. Corrected meteorological data was submitted in an August 1990 FSAR amendment. The Chapter 15 analysis will be revised by October 1991 to address the changed X/Q values.

2.6.4.2
PER 289-0192

Diving inspections of two Service Water (SW) System pipe supports revealed accelerated corrosion degradation on approximately one-third of the hex-nuts applied to the baseplate concrete anchor studs. The baseplates are submerged within the 1B service water spray pond and are associated with hanger mark numbers SW-936N and SW-937N. The concrete anchor studs as well as the coated support steel did not show any appreciable corrosion degradation.

Under water repair actions were successfully completed with hex-nut materials selected for optimum corrosion performance when combined with newly installed (enhanced) cathodic protection elements. Torquing requirements at all concrete anchor bolt locations were satisfied and it was concluded that both SW-936N and SW-937N were restored to their original design integrity.

Safety Evaluation Summary

In the interim, prior to the repair effort, a full deadweight and seismic analysis of the service water piping was completed assuming that the corrosion affected baseplates were out of service (i.e., they retained zero load carrying capability). These analyses demonstrated that, in spite of the conservatively assumed degraded piping support condition, ASME Code piping stress limits were still satisfied. In addition, the system was stable under the slightly-increased seismic deflections. These results constituted the basis of a Justification for Continued

Operation (JCO) and the conclusion that an Unreviewed Safety Question did not exist since 1) no service water operational function was impaired, 2) design basis pressure boundary stress limits were not exceeded under worst case assumptions of baseplate integrity, and 3) no Technical Specification margin of safety was reduced. Finally, it should be noted that, following discovery of the situation, repairs to the supports were completed in a timely manner (approximately a three week period) by a contract diving team. The probability of a major seismic event occurring in this brief window is remote. The plant was also in a shutdown condition during the course of the repair efforts.

2.6.4.3
<u>PER 290-0399</u>

A Justification for Continued Operation (JCO) was written to support the condition of the emergency Control Room Chillers. One division was operable and one division was inoperable for maintenance and replacement of parts. The redundant unit was unavailable to provide Emergency Control Room Cooling in the emergency condition. The emergency Control Room Chiller design basis is to provide additional cooling capacity for personnel comfort such that the temperature is less than 85°F in the Control Room. The Standby Service Water system alone can provide sufficient cooling for equipment operability with Control Room temperatures below 104°F.

Plant Procedures were deviated to incorporate instructions to reduce Control Room lighting to limit heat generation at the operators' discretion when the temperature is between 85° and 104°F Post-accident.

<u>Safety Evaluation Summary</u>

There is no Technical Specification which addresses the availability of the Emergency Control Room Chillers. The evaluation demonstrated that the unavailability of one Emergency Control Room Chiller does not present a hazard to public safety, or to the safety of the Plant personnel.

The unavailability of one Emergency Control Room Chiller did not result in a change to the WNP-2 Technical Specifications, and the Unreviewed Safety Question evaluation concluded: The probability or consequences of a new accident is not increased, no new accident or malfunction could be introduced, and the margin of safety for control room personnel performance is not affected.

2.6.4.4
<u>PER 90-477</u>

During the annual refueling outage, the breaker (E-CB-73/7A) between the Low Voltage Critical Switchgear (E-SL-73) and Motor Control Center 7A (E-MC-7A) tripped causing a loss of power to several loads including Reactor Protection System (RPS) Bus A. Loss of power to RPS Bus A caused a half-scram in RPS Division A and multiple primary containment isolations, which are Engineered Safety Feature (ESF) actuations (See LER 90-013 for further details).

The root cause evaluation required a test of the breaker and no Quality Class I spare breakers were available as a replacement. The decision was made to install a Quality Class II breaker in the circuit while the problem breaker (E-CB-73/7A) was tested.

Safety Evaluation Summary

The Safety Evaluation directed that all loads fed by breaker 7-73 be considered inoperable during the presence of the Quality Class II breaker. A complete divisional outage is allowed with the plant in Modes 4 or 5, which was the case during the installation of the temporary breaker. Therefore, this activity did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question.

2.6.4.5
PER 290-533

An AC voltage drop test on the field windings for division II Diesel Generator (DG-GEN-DG2) was performed in June 1990. This test revealed that the windings for pole no. 6 had shorted turns.

DG-GEN-DG2 has eight field poles on the rotor which provide field flux required to generate voltage in the stator of the generator. An AC voltage drop test was performed by applying 120vac across two opposite poles in series. Then the voltage across each pole was measured. For good windings, the voltage drop across each pole should be equal (60v each). When 120v ac was applied across pole no. 2 and pole no. 6 in series, the voltage drop across these poles was measured to be 84v and 36v respectively. This indicated that some of the winding turns on pole no. 6 were shorted.

Safety Evaluation Summary

A Safety Evaluation was performed to justify continued operation with the above condition on DG-GEN-DG2. The evaluation showed that shorted turns could have the following effects on the generator performance:

i) Reduce the magnetic flux produced by the pole and; hence, reduce the generator output voltage for a given field current.

ii) Increase heating within the pole winding and degrade it with time.

iii) Unbalance the magnetic field across the air gap, causing vibration.

Following the discovery of the shorted turns, the unit was run at 100% for 72 hours. During this test the field current at full load was 142 amps which is less than the rated current of 168 amps for the voltage regulator. The AC voltage drop test was repeated after the run and there was no change in the test results. Also, the surveillances were reviewed and no excessive vibration was noted.

A new procedure was implemented to periodically perform the AC voltage drop test. The latest test was performed on January 28, 1991. Results of these tests did not show any noticeable degradation of pole windings.

From the tests performed to-date, it is clear that the unit is capable to perform its safety function (i.e., to supply the Division II loads on loss of offsite power).

The review of tests performed on the unit concluded that (1) the performance of the unit 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.

2.6.4.6
PER 90-523

This Problem Evaluation Report address the problem of high failure rate on Rosemont 1153 Series B and D pressure Transmitters due to loss of fill-oil. NRC Notice 90-01 required a documented basis for continued operation for transmitters of this type. Ten of these devices are installed in the plant.

Safety Evaluation Summary

The earliest symptom a model 1153 Series B or D transmitter will exhibit prior to failure is that, if it is leaking fill-oil, a sustained drift will be observed. The calibration data for the installed transmitters were reviewed and found to show no drift trends indicative of a fill-oil loss failure. Drift data was found to be within the Rosemount limits for response time degradation. The existing surveillance monitoring will be continued, with transmitter calibration data analyzed for sustained drift. This resolution is considered a short term solution until the transmitters are replaced.

2.6.4.7
PER 290-0734
PPM 4.602.A6
EOP Flow Chart

This Problem Evaluation Report described a problem with the circuit breaker (CB-RPT-3A and 3B) and switch design associated with the Reactor Recirculation (RRC) Pumps. Breakers CB-RPT-3A and 3B were found to not have a seal-in function that seals in the trip circuit after a trip signal is received. If the ATWS signal clears, reclosing the tripping circuit before RRC pump speed decays below the low speed pickup point, the pump would re-energize and continue to run at low speed.

Changes were implemented in appropriate procedures to prevent the possibility of the inadvertent restart, at 15 Hz operation, of the RRC pumps after a pump trip from 60 Hz operation is initiated by the ATWS RPT signal. This was done by using the "pull to lock" function in the CB-RPT-3A and 3B switches, which prevents the energization of the 15 Hz power circuit even if all permissives are in the enable condition.

Safety Evaluation Summary

The Safety Evaluation showed that neither a Technical Specification change was required nor did an Unreviewed Safety Question exist in the application of the Emergency Operating Procedure (EOP) to the WNP-2 ATWS analyses. The study showed that the re-energization event resulted in only a comparatively small quantity of energy being generated; that is, even if the RRC pumps experience a restart after the ATWS signal clears, the effects of the energy produced are bounded by the results of the full power analysis conducted as a part of the same program. The "pull to lock" function is, therefore, a redundant action that enhances the mitigating responses to the ATWS transient. Moreover, the "pull to lock" function does not adversely impact other transients or functions of the RRC Low Frequency Motor Generator (LFMG) mode.

2.6.4.8
PER 290-0786

The Problem Evaluation Report was written because the fatigue analysis for the Reactor Core Isolation Cooling (RCIC) head spray piping and associated RPV nozzle did not reflect actual plant practice. Actual use of RCIC is more frequent than was accounted for in the original analyses. Also, RCIC use occurs during "shutdowns" and "other scram" situations during which RCIC use was not originally anticipated.

There was no change to the physical plant. A Justification For Continued Operation (JCO) was prepared and approved. The WNP-2 specifications and related calculations were revised to reflect actual plant use. Reanalysis by GE of the RCIC Head Spray nozzle is required and engineering services for this effort are being procured.

Safety Evaluation Summary

The additional use of RCIC was reviewed and it was determined that sufficient margin exists in the RCIC head spray piping and the reactor vessel nozzle designs to accommodate current WNP-2 operating practices. The number of system operational cycles are considered in the ASME code-required fatigue analysis of two different pressure boundary analysis for the RCIC head spray. The first analysis is the Architect/Engineer's analysis of the Head Spray piping. The other analysis is the GE evaluation of the RPV RCIC Head Spray nozzle.

2.6.4.9
PER 290-831

This Problem Evaluation Request described a potentially nonconservative Primary Containment Instrumentation Piping Thermal Analysis. The instrumentation piping attached to process lines within primary containment were analyzed (over their full length) by the Architect Engineer at the associated process line operating temperature. These instrument lines are uninsulated, dead-ended services and, as such, they carry no process flow. As a result, the instrument lines only maintain the process line temperature within ten to fifteen pipe diameters of their attachment point to the process line. The balance of the line is, thus, in equilibrium with the containment ambient temperature. All of the subject instrument lines are 3/4 or one (1) inch stainless steel pipe runs.

Safety Evaluation Summary

The described elevated instrumentation piping thermal analysis fault is conservative (i.e. it over predicts stresses) in many cases. However, cases do exist where the excess instrument line thermal expansion works to reduce piping thermal stress states by achieving a false net zero thermal anchor movement between the large bore process piping and the attached instrument piping. Fortunately, the small diameter instrument piping has a high degree of flexibility and can accommodate significant displacements without developing thermal expansion stress responses which exceed ASME Code limits. As an example, a worst case configured instrument system (designated X42a) was reanalyzed and it was shown that the increased thermal stresses were still in compliance with ASME limits. As a followup, a survey of all inside containment instrument piping anchor groups was completed and the susceptible cases were identified for thermal reanalysis and full requalification in compliance with ASME Code requirements. Where needed, support adjustments will be completed at the next maintenance outage to assure that good design practices for thermal expansion are adhered to in all WNP-2 piping installations.

A JCO was prepared based on the cited analysis results of instrument system X42a, a worst case configuration. Based on these analyses it was concluded that an Unreviewed Safety Question did not exist since 1) no instrumentation function was impaired, 2) design basis pressure boundary stresses met the requirements of the governing ASME Code, and 3) no Technical Specification margin of safety was reduced.

2.6.4.10
PER 90-835

This Problem Evaluation Request described a problem with the LOCA accident analysis supporting the 9 X 9 fuel. It had been determined that the current fuel vendor had not accounted for the time required to accelerate the HPCS pump from de-energized to rated conditions.

Safety Evaluation Summary

A Justification For Continued Operation (JCO) was written which showed that, by changing the initiation signal for the HPCS start, the timing used in the analysis was acceptable. The fuel vendor, Advanced Nuclear Fuels (ANF), assumed flow to commence within 18.5 seconds after the start of the LOCA, and for the HPCS system to receive a start signal on low water level at 7.5 seconds into the LOCA. By taking credit for high drywell pressure as the initiation signal, the HPCS system receives the initiation in less than one second. This allows greater than 5 seconds for the HPCS pump to achieve rated speed, making the 18.5 second assumption bounding. Further, the LOCA analysis had approximately 450°F margin to the 10CFR50.46 peak cladding temperature (PCT) requirements. This change did not constitute a change to Technical Specifications in that HPCS was always operable, and there would be no change required to the APLHGR curves. These are the only items germane to this subject in the plant Technical Specifications.

2.6.5 PLANT TESTS AND EXPERIMENTS

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

2.6.5.1
Temporary Procedure 8.4.60

Governor valves used to control steam flow to the main turbine are controlled by DEH (a Digital Electro-Hydraulic control system). In order to assess the impact of governor valve position on plant thermal efficiency while at near 100% reactor power, one of the two partially-open governor valves was placed in valve test. In a controlled manner, the two governor valves were repositioned with one ~10% open and the other ~90% open. To preclude any uncertainty about DEH response in the event that a bypass valve opened causing a DEH mode change [changing from mode 4 (Turbine Follow Reactor Manual Mode) to mode 3 (Turbine Load Control Mode)] while the governor valves were in test, a temporary jumper was installed.

Safety Evaluation Summary

The functionality of the DEH control system with the jumper installed and with the governor valves in an optimized configuration did not result in a change to WNP-2 Technical Specifications or involve an Unreviewed Safety Question because (1) the function of the DEH control system in mode 4 did not change, (2) the margin of safety provided in the Technical Specifications was not changed, and (3) the boundary conditions for the FSAR were not changed.

2.6.5.2
Temporary Procedure 2.8.14

This temporary procedure provided direction for partial draining of the Spent Fuel Storage Pool. The Spent Fuel Pool B diffuser check valve, FPC-V-146B, appeared to be stuck shut. The valve is located in one of the two diffuser lines that supplies the Spent Fuel Pool. The valve is approximately 19 inches below the water line of the Pool. The valve could not be isolated from the Spent Fuel Pool, so the level was lowered below that of the valve to perform the maintenance.

Safety Evaluation Summary

The temporary procedure provided the cautions and directions to lower the pool level in a controlled manner. The procedure required entrance into the action statement for Technical Specification 3.9.9, which prohibits fuel handling or the transfer of any load over the pool while the pool level was lowered. Therefore, the basis for the specification, prevention of spent fuel damage, was maintained.

2.6.5.3
Temporary Procedure 8.3.203

This temporary procedure was performed to determine the operability of valve MS-V-146. The valve was not stroke tested following maintenance on the valve stem packing. Without post-maintenance testing, the valve could not be considered operable. When the lack of testing was discovered the valve was declared inoperable and the Technical Specification action statement (L.C.O. 3.6.1.4) for the loss of one MSLC subsystem was entered. With the plant at power the valve could not be fully closed without causing a plant shutdown. This procedure was written to partially close the valve and take a current signature during the stroke to determine if the valve motor current remained within acceptable limits.

Safety Evaluation Summary

If MS-V-146 were to close it would result in isolation of the Main Steam Bypass Valves located downstream. A generator load rejection with bypass valve failure is an analyzed accident and has a failure probability in the moderate frequency range. To increase the probability of this accident during this test would require either the valve MS-V-146 going full closed, thus isolating bypass valve capability, or having reduced bypass capacity while the valve is stroked partially closed.

There was no discernable increase in probability for failing the valve MS-V-146 closed. The procedure had a precaution/limitation that required opening the electrical disconnect switch if the valve moved more than 15 seconds. During performance of the procedure, personnel were stationed at the MCC cubicle and were in contact with the control room. During installation of the switch, all work was "second verified" and the operation of the switch was tested during the installation. To close the valve inadvertently would require the simultaneous failure of the switch and the electrical disconnect which is not credible. There is no increase in the failure frequency from "moderate" to "normal."

The testing of valve MS-V-146 did not increase the consequences of any analyzed accidents. Generator load reject with bypass failure already bounds the worst-case failure during this test of fully closing MS-V-146. There were no off-site dose consequences from the generator load reject with bypass failure.

2.6.5.4
Temporary Procedure 8.3.111

This temporary procedure was written and performed to investigate operation of the Reactor Water Cleanup (RWCU) System at increased flow rates. The flow rate was increased to 183 percent of normal to allow evaluation of system performance under controlled conditions. The results of this test were used to establish a routine higher flowrate in the RWCU system to improve reactor water chemistry.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve an Unreviewed Safety Question because (1) the margin of safety in the Technical Specifications was not reduced by the temporary increase in RWCU flow rate, and (2) the boundary conditions of the FSAR evaluations were not impacted by the flow increases.

2.6.6 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 involve an Unreviewed Safety Question. Plant procedure changes associated with other change documents such as Plant Modifications or Lifted Leads and Jumpers are described in other subsections of this report. The following are summaries of significant Plant Procedure changes not covered elsewhere in this report that were processed during 1990:

2.6.6.1
PPM 7.0.2
PPM 7.4.8.1.1.2.1
PPM 7.4.8.1.1.2.2

Plant Procedures were modified to ensure Diesel Generator operability under accident conditions that could result in elevated temperatures in the Diesel Electrical Equipment Rooms. A recalculation had found that the Emergency Diesel Generator Static Exciter Voltage Regulator (SEVR) would not function in the high temperatures found in the equipment rooms (see LER 90-020 for additional details). The revised plant procedures call for Plant Operators to remove SEVR subcompartment doors to limit the temperature rise when room temperatures exceed predetermined limits.

Safety Evaluation Summary

The safety evaluation concluded the DG operating margins were maintained and the basis for the Technical Specifications had not changed. An Unreviewed Safety Question did not exist since the procedure changes allow the SEVR (and therefore, the Diesel Generators) to operate satisfactorily within the design basis.

2.6.6.2
PDF 90-1023 and 1025
ISCR 1021
PER 90-750

This procedure and instrument setpoint change was made to match Reactor Water Cleanup (RWCU) Flow Transmitter 41 (RWCU-FT-41) to its flow element orientation. The change used an empirically-based calibration curve to match RWCU-FT-41 with the flow element.

Safety Evaluation Summary

This change did not result in a change to the WNP-2 Technical Specifications or involve a Unreviewed Safety Question because, (1) the change did not result in a loss of isolation capability for the RWCU system, and (2) this instrument is not used to support the high energy line break safety analysis.

2.6.2.3
PPM 2.5.7, 3.1.2, and 3.1.3

These changes to plant procedures were initiated as a result of GE SIL 502 (OER 89075), which identified potential violations to the critical power ratio during a single turbine valve slow closure transient. The changes raised the normal DEH flow limiter setting from 110% to 130% in order to be above the GE-recommended minimum of 115.5% and provide increased operational reliability.

Safety Evaluation Summary

This did not require a change to the WNP-2 Technical Specifications or result in an Unreviewed Safety Question because the new setting resulted in only a minor change to a non-limiting transient (Pressure Regulator Failure-Open, FSAR 15.1.3). The sequence of events and; hence, the consequences of this type of malfunction are changed. The G.E. Transient Safety Analysis Design Report for WNP-2 (GEZ-6413) indicates that a DEH flow limiter setting of 130% will result in a turbine trip due to level·swell as a result of a pressure regulator failure. A DEH flow limiter setting of 110% yields an MSIV isolation as a result of a pressure regulator failure. This change in the sequence of events is bounded by the turbine trip/generator load reject transient which is considered separately. Therefore, the consequences of this change in DEH flow limiter setting are not considered to be increased.

2.6.6.4
PPM 4.602.A5
ISCRs 985 and 986

This procedure change and Instrument Setpoint Change Requests 985 and 986 revised the high radiation alarm setpoints for two Area Radiation Monitors (ARM-RIS-24 and ARM-RIS-5), to compensate for increased background radiation levels. Background radiation levels increased due to fuel handling operations during Refueling Outage R5. Alarm setpoints are selected to provide indication of any abnormal increase in radiation levels, while minimizing false alarms. The setpoints were subsequently lowered post-outage reflecting decreased background levels.

Safety Evaluation Summary

These instruments are not specifically addressed in the WNP-2 Technical Specifications. ARM alarm setpoints are dependant on background radiation levels. Periodic adjustments for changes in background due to changes in operating conditions or special evolutions does not constitute a modification or change that has the potential for reducing any safety margins, nor increase or create the possibility or probability of occurrence of an accident or malfunction.

2.6.7 FIRE PROTECTION PROGRAM CHANGES

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

2.6.7.1 SCN 89-029

SCN 89-029 was developed to incorporate into the FSAR those fire protection requirements which were removed from the Technical Specifications in Amendment 67. The SCN incorporates minor changes to the fire protection program as follows:

1. This change (paragraph F.5.2.3.d) revises the Technical Specification 4.7.6.1.1.d requirement that the fire protection water system be demonstrate operable "at least once every six months by performance of a system flush". The SCN requires that the main fire header be flushed annually.

   The Technical Specifications defined the fire protection water system as water supply, pumps, and distribution piping to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each deluge or spray system required to be operable in accordance with Specifications 3.7.6.2, 3.7.6.4 and 3.7.6.5. The implementation of the Technical Specification requirements to perform a system flush every six months would; therefore, ensure that the water flow path to each required sprinkler system, water spray system, hydrant valve, or hose standpipe was free of obstruction every six months.

   The SCN, through requirements for ring header flush, hydrant flow tests, sprinkler system testing and hose station flow tests ensures that the testable sections of the fire water system are clear. The lead-ins to the sprinkler systems are not flushed (main drain tests) to reduce the potential for radioactive contamination during testing (due to the large quantity of water discharge with no connection to plant drain system). The SCN reduces the frequency of flushing to an annual basis, based on recent testing which showed minimal debris.

2. Technical Specification 4.7.6.1.2.c required a diesel inspection at least once per 18 months. SCN paragraph F.5.2.3.2.c specifies that the diesel be inspected in accordance with the diesel manufacturer's recommendations. This change avoids unnecessary tear-downs of the diesel, as the manufacturer does not recommend inspection every 18 months.

3. Technical Specification 3.7.6.2 required a continuous fire watch in the event that a spray/sprinkler system which protects redundant systems or components is inoperable. The SCN paragraph F.5.3.2.a specifies the systems which protect redundant systems or components (systems #65, #66). Because these two systems are pre-action systems, the SCN allows the pre-action system to be operated as a wet-pipe system to provide suppression coverage in lieu of posting a continuous fire watch, until the system is restored to full operability as a pre-action system.

4. SCN paragraph F.5.2.3 does not include Technical Specification 4.7.6.1.2.a.2 statement that fire pumps be tested "on recirculation flow". This change in wording is not significant.

5. SCN paragraph F.5.7.3.2 states that fire doors will be verified closed daily during routine operator plant tours and inspected weekly to verify that the doors are not damaged or obstructed. Removed Technical Specification 4.7.7.2 required unlocked fire doors without electrical supervision to be verified closed daily. Reference to testing of electrical supervision of the doors is removed as this testing is not applicable to fire doors at WNP-2.

6. SCN paragraph F.5.7.3.1.d has been added to allow to clarify the inspection requirements for certain access doors not normally used for personnel traffic. The access "doors" are often classed as penetrations and inspected under the requirements of removed Technical Specification 4.7.7.1. Therefore, for consistency with the removed Technical Specifications, the SCN requires these doors to be inspected at least once per 18 months.

7. SCN paragraph F.5.6.2 differs from removed Technical Specification 3.7.6.5 in that the Technical Specification required that backup fire hose be provided within 1 hour if the hydrant provides the primary means of fire suppression. The SCN allows 24 hours to provide the backup hose. This change was based on the inventory of fire hose in the Fire Brigade Van.

8. Removed Technical Specification 4.7.6.5.b specified the spring/fall months for a six-month hydrant inspections. SCN paragraph F.5.6.3.b specifies hydrant barrel should be verified "drained", and the removed Technical Specification required that the hydrant barrel be verified "dry". This change is not significant.

9. SCN paragraph F.5.5.2 differs from removed Technical Specification 3.7.6.4 in that the removed Technical Specification allows 24 hours to install compensatory hose in any area protected by fixed fire suppression system. The SCN specifies 1 hour for installation of compensatory hose for all essential hose stations. This change is more conservative than removed Technical Specification requirements.

10. SCN paragraph F.5.4.3.c.1 differs from removed Technical Specification 4.7.6.3.c.1 in that the SCN does not specify verification of actuation of associated closure devices. As the PGCC halon systems do not include any associated closure devices, this change does not appear significant.

11. SCN paragraph F.5.8.2 differs from removed Technical Specification 3.3.7.9 in that the removed Technical Specification action statement allowed different compensatory actions based on the function of the inoperable detector (early warning or used to actuate a fire suppression system). Under certain conditions, the removed Technical Specification allowed the inoperable detectors to be restored to service within 14 days <u>or</u> a fire tour be established within 1 hour. For simplicity, the SCN does not distinguish between the function of the different detectors, but requires a fire tour to be established within 1 hour if any of the required detectors is out of service. The removed Technical Specification option to not post a fire tour if the detector is restored to service within a certain time period is not included in the SCN as an option. The removed Technical Specification basis has been modified to reflect this program change and is incorporated in the SCN in paragraph F.5.1.1.

12. SCN table F.5-3 differs from removed Technical Specification.7.7.9 in that the SCN does not specify the type or number of detectors installed in each plant area and does not distinguish between early warning and detectors which actuate fire suppression systems. The type, number, and function of each detector was included in the removed Technical Specifications as the Technical Specification allowed different compensatory measures based on the detector function. The SCN requires the compensatory measures to be initiated when <u>any</u> detector within an essential fire detection zone is inoperable.

13. SCN paragraph F.5.8.3 differs from removed Technical Specification 4.3.7.9. The removed Technical Specification required all detectors be verified operable by performance of a channel functional test at least once per six months. The SCN modifies the frequency of testing for smoke detectors to require visual inspection every six months and a channel functional test every 12 months. Similarly, normally inaccessible detectors are tested during cold shutdown exceeding 24 hours unless tested in the previous 12 months. The NFPA 72D detector supervision circuits are verified operable during the channel functional test of the detector zone.. The change in the frequency of the channel functional tests for smoke detectors was previously resolved.

14. Technical Specification 4.7.6.3 required the performance of a flow test through accessible headers and nozzles of the PGCC halon systems every 18 months to assure no blockage. SCN paragraph.5.4.3.e requires the performance of this test every five years based on the following:

    a. The PGCC area is not readily accessible; therefore, the halon nozzles are not subject to inadvertent damage or plugging.

    b. The reduction of the test frequency reduces the challenges to the system caused by the complexity of the test and reset procedure.

2.6.7.2
<u>SCN 90-024</u>

This SCN revised the fire door inspections from daily to weekly and makes provisions for not performing the inspections if plant conditions (Such as ALARA) cause a conflict.

This change does not increase the probability of occurrence of an accident or malfunction of equipment important to safety, as the change does not modify plant equipment. The change affects only the procedural means to ensure that unlocked fire doors are checked shut daily.

2.6.7.3

The frequency for testing of the high temperature alarms associated with the manually actuated suppression systems on the Standby Gas Treatment (SGT) System charcoal filter beds (Fire Protection Procedures 15.2.34 and 15.2.35) was changed from semi-annual to refueling outage.

2.6.7.4

The fire door routine test procedure, PPM 15.1.2, was modified to add an annual inspection of fire door clearances and latching mechanisms. Although not specifically required by the fire protection program, this change was initiated to provide assurance that the fire doors continue to conform to the installation requirements of NFPA 80, "Standard for Fire Doors and Windows".

2.6.7.5
<u>LLJ 90-184</u>

A temporary change was made to pressurize the Fire Protection System with the Potable water system while the Circulating Water Basin was drained during the annual refueling outage. The cross-tie provided added assurance that the fire protection needs of the plant would be met.

2.7 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 Two Emergency Diesel Generator (DG-2)
   January 7, 1990

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

   This was the Second Failure of the last 100 valid tests.

3. Cause of failure:

   During performance of the Monthly Surveillance Test, the Division Two Diesel Generator tripped on Engine Overspeed. An Investigation revealed that the Limit Switch Linkage Arm Set Screw was found to be loose. This limit switch monitors the position of the Overspeed Reset Lever, and initiates a trip of both engines when upward movement of either reset lever is detected.

4. Corrective measures taken:

   The limit Switch Linkage Arm was readjusted to the proper position, consistent with the other Engine Devices. The Allen Set Screw was then tightened to maintain this linkage position.

5. Length of time diesel generator unit was unavailable:

   The Diesel Generator was out of service for 15 1/2 hours, and returned to service at 1810 Hrs on Sunday January 7, 1991.

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.

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

   Division Three Emergency Diesel Generator (DG-3)
   May 23, 1990

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

   This was the Second Failure of the last 100 valid tests.

3. Cause of failure:

   While performing the HPCS DG LOCA Test per PPM 7.4.8.1.1.2.8, at step C.5) the HPCS Diesel Generator exhibited erratic frequency and voltage control. When the operator tried to raise the load, the frequency and the voltage both started oscillating. The High Pressure Core Spray Pump, HPCS-P-1 was tripped off line and then the HPCS Diesel Generator was tripped. Troubleshooting of the Engine controls revealed cold solder joints on the Woodward governor control board.

4. Corrective measures taken:

   All Woodward Governor controls on the other units were inspected as well as the new unit installed on the HPCS Diesel. Cold solder connections were also found on the new control board that was to be installed on the HPCS Diesel, and were repaired prior to installation. No other suspect solder connections were found on any control board inspected.

5. Length of time diesel generator unit was unavailable:

   The HPCS Diesel Generator was out of service for 9 days and returned to operable status on June 1, 1991 at 0640.

6. Current surveillance test interval:

   The testing frequency was accelerated to once every 7 days. This accelerated testing continued for 20 consecutive tests, ending on October 26, 1990.

7. Verification of test interval:

   The surveillance test interval of seven days is in conformance with WNP-2 Technical Specifications Table Number 4.8.1.1.2-1.