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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

Docket No. 50-397

February 28, 1989

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

Dear Mr. Martin:

Subject: NUCLEAR PLANT NO. 2 ANNUAL REPORT

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

2) WNP-2 Technical Specifications, 6.9.1.4 and 6.9.1.5

3) Regulatory Guide 1.16, Reporting of Operating Information - Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Report for calendar year 1988. Should you have any questions or comments please contact M.R. Wuestefeld, WNP-2 Assistant Plant Technical Manager.

Very truly yours,

JWBaker

C.M. Powers
Plant Manager

CMP:MRW:TRW

Attachments

IE56
1/1

ANNUAL OPERATING REPORT

OF

WNP-2

FOR 1988

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

8903130595

TABLE OF CONTENTS

1.0 INTRODUCTION

The 1988 Annual Operating Report of Washington Public Power Supply System Plant Number 2 (WNP-2) is provided as a supplement to the Monthly Operation Report. This report is submitted in accordance with the requirements of Federal Regulations and Facility Operating License NPF-21. It should be noted that, for ease of reference and completeness, additional required reports are also included. WNP-2 is a 3323 MWt, BWR-5, which began commercial operation on December 13, 1984.

On January 18, 1988 the Plant was shutdown after 35 days of operation to correct a condenser tube in-leakage problem. Several leaking tubes were plugged and the Plant was returned to service. Due to condenser in-leakage and rapidly increasing conductivity the Plant was again shutdown on February 13, 1988 for condenser tube and baffle plate repairs. During this shutdown, the reactor building (secondary containment) was overpressurized by the inadvertent start of a reactor building air supply fan, causing the designed roof rupture panels to relieve with resultant damage to the roof. As a result of Engineering analyses/evaluations, and inspection and testing performed, it was determined that the roof ruptured as designed. In addition, no other damage was found and the repair effort successfully restored secondary containment to an operational condition.

From April 30, 1988 to June 19, 1988 the Plant was in a shutdown condition as scheduled for the annual maintenance and refueling outage. Following the outage, the Plant was restarted and operated until August 24, 1988 when it was shutdown because of unidentified leakage in primary containment which exceeded Technical Specification limits. Reactor Core Isolation Cooling (RCIC) valve RCIC-V-63 was identified as the major source of leakage and the valve was repaired and returned to service.

On September 11, 1988 reactor power was reduced to approximately 6% to allow drywell entry for inspection of Containment Supply Purge System piping welds. The inspection was required after liquid nitrogen cracked a welded joint in a six-inch pipe outside of containment while Plant Operators were inerting containment.

A forced outage occurred on October 27, 1988 due to a cracked weld in a main steam line trap station drain in the Turbine Building. Repairs were made and the Plant was returned to service within 48 hours.

On December 1, 1988 the Plant was shutdown again due to air leakage through Containment Supply Purge Valves CSP-V-9 and CSP-V-5 which exceeded Technical Specification limits. The valve seats on the CSP valves were replaced and the Plant was restarted and ran at or near 100% capacity for the remainder of the year.

During 1988, there were several examples of major accomplishments which required significant effort on the part of Supply System personnel to successfully complete. The following is a summary of those efforts:

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

- o Replacement of both Reactor Water Cleanup (RWCU) pumps. The new pumps, which are unique because both the motor and pump are combined in one unit, do not have seals and are designed for zero leakage.

- o High pressure turbine inspection. The turbine was dismantled, cleaned and inspected to ensure there were no cracks or defects in the turbine blades.

- o Installation of the Anticipated Transient Without Scram-Alternate Rod Insertion (ATWS-ARI) System. As an independent system to the Reactor Protection System (RPS), the ATWS-ARI System acts as a back-up to ensure that the control rods are inserted when an RPS actuation occurs.

- o Overhaul of the Diesel Generators. Two of the five diesel engines were overhauled. Modifications were installed on the Division I and II diesel engines to allow idle speed operation during surveillance testing.

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

(b) WNP-2 continued to have an excellent record for limiting worker radiation exposure. In 1988, total radiation exposure at the Plant was 352 man-rem. The Institute for Nuclear Operation (INPO) has set 460 man-rem as the industry goal for 1990 for BWRs.

(c) During the year WNP-2 experienced only one unplanned automatic shutdown (Scram), which is well below the industry average of 2.7. The reactor automatically shutdown on February 4, 1988 from 100 percent power due to Main Steam Isolation Valve isolation caused by improper execution of a Technical Specification surveillance procedure.

(d) In terms of electrical output, WNP-2 provided more than six billion kilowatt-hours to the Bonneville Power Administration. This amount marked an 11 percent increase in electrical generation over 1987. In addition, the Plant was available for power production nearly 69 percent of the time during 1988, exceeding the overall industry average for BWRs of 65 percent.

(e) Another new generation mark was established in November, when a net 768,651,000 kilowatt-hours was generated (the highest for any 30-day month in the history of the Plant).

During the year WNP-2 received 20 NRC Notice of Violations (NOVs): Seventeen (17) Level IV and three (3) Level V.

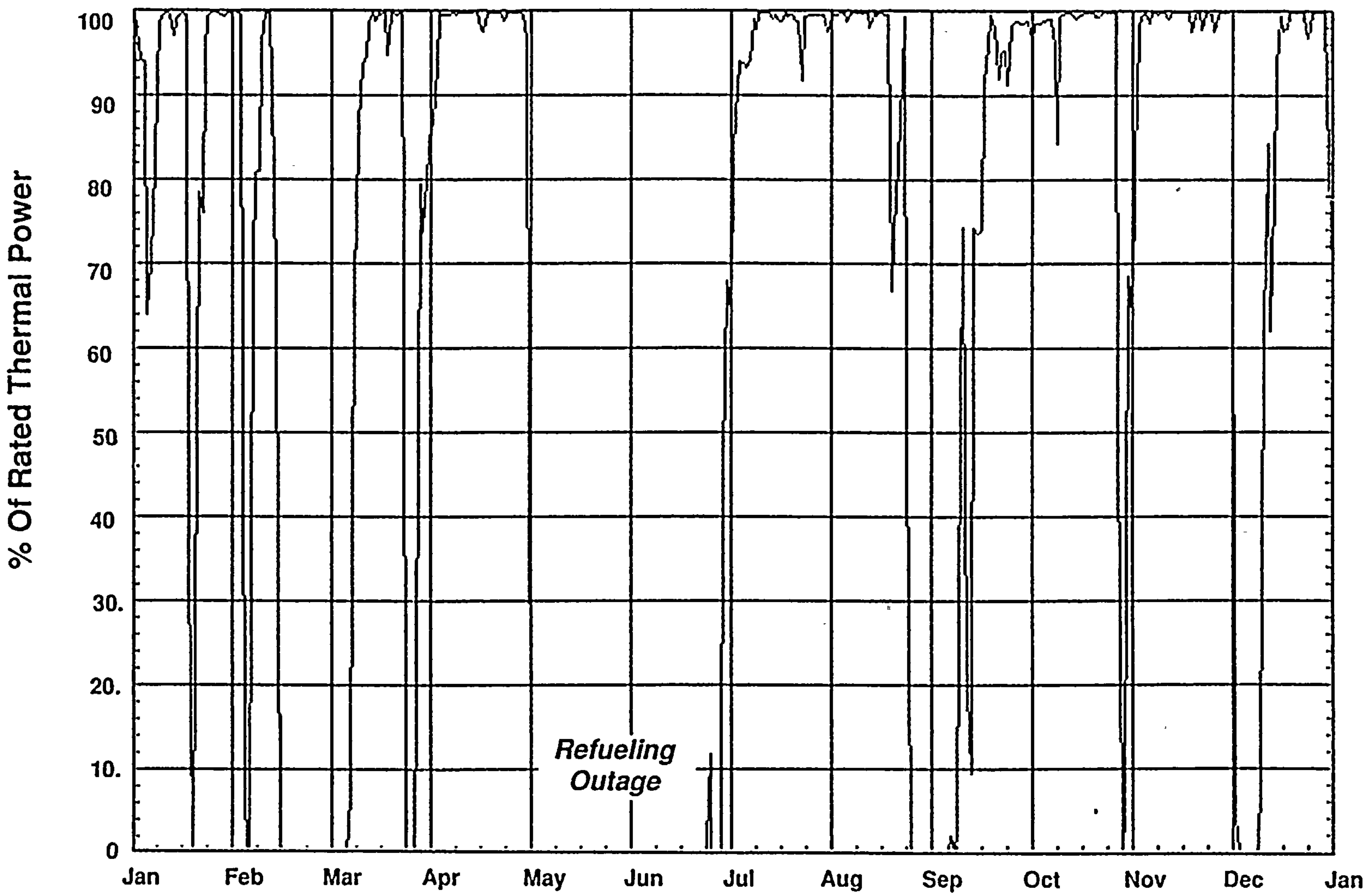
Also during 1988, a total of 38 License Event Reports (LERs) were written and submitted pursuant to the requirements of 10CFR50.73.

The 1988 capacity factors, based upon net electrical energy output, are listed in the following table.

Month	Capacity Factor
January	85.7
February	33.6
March	60.3
April *	91.6
May	0
June **	5.8
July	93.9
August	70.6
September	58.3
October	86.1
November	97.5
December	63.5
Overall	62.2

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

WNP-2
1988 Power History



1988

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

890049.1

2.0 REPORTS

The reports provided in this section meet the requirements of Federal Regulations (10CFR50.59) and the WNP-2 Operating License. Complete data for the year 1987 has been included.

NUCLEAR PLANT NO. 2

REPORT FOR CALENDAR YEAR 1988

		NUMBER OF PERSONS RECEIVING OVER 100 MREM			TOTAL MAN-REM		
		STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS	STATION EMPLOYEES	UTILITY EMPLOYEES	CONTRACTORS AND OTHERS
OPERATIONS & SURVEILLANCE	MAINTENANCE PERSONNEL	20.539	0.000	1.024	9.806	0.000	0.125
	OPERATING PERSONNEL	30.306	0.000	0.000	25.746	0.000	0.000
	HEALTH PHYSICS PERSONNEL	25.908	0.000	12.663	25.451	0.000	5.280
	SUPERVISORY PERSONNEL	12.369	0.077	0.397	6.947	0.030	0.130
	ENGINEERING PERSONNEL	6.816	7.740	3.191	2.321	2.118	0.935
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	82.894	0.090	61.034	44.517	0.067	20.239
	OPERATING PERSONNEL	3.516	0.000	0.000	2.817	0.000	0.000
	HEALTH PHYSICS PERSONNEL	4.175	0.000	5.069	7.233	0.000	3.712
	SUPERVISORY PERSONNEL	0.830	0.461	0.749	0.358	0.421	0.143
	ENGINEERING PERSONNEL	5.529	7.659	5.569	1.445	3.230	2.124
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	1.265	0.000	14.011	0.895	0.000	5.210
	OPERATING PERSONNEL	0.326	0.000	0.000	0.288	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.803	0.000	0.059	1.283	0.000	0.045
	SUPERVISORY PERSONNEL	1.133	0.446	0.124	0.298	0.349	0.040
	ENGINEERING PERSONNEL	0.366	2.533	5.275	0.141	1.107	0.900
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	107.741	0.407	79.906	79.180	0.219	30.556
	OPERATING PERSONNEL	2.428	0.000	0.000	2.145	0.000	0.000
	HEALTH PHYSICS PERSONNEL	9.627	0.000	17.624	15.739	0.000	11.098
	SUPERVISORY PERSONNEL	2.863	1.016	1.730	0.970	0.461	0.452
	ENGINEERING PERSONNEL	12.787	9.804	12.551	3.394	3.580	6.117
WASTE PROCESSING	MAINTENANCE PERSONNEL	1.515	0.000	0.000	1.152	0.000	0.000
	OPERATING PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.813	0.000	0.696	1.092	0.000	2.820
	SUPERVISORY PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
	ENGINEERING PERSONNEL	0.000	0.000	0.000	0.000	0.000	0.000
REFUELING	MAINTENANCE PERSONNEL	7.925	0.008	2.023	7.084	0.004	0.277
	OPERATING PERSONNEL	0.699	0.000	0.000	0.595	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.376	0.000	3.536	0.575	0.000	1.168
	SUPERVISORY PERSONNEL	0.679	0.000	0.000	0.644	0.000	0.000
	ENGINEERING PERSONNEL	0.901	0.239	0.158	0.260	0.052	0.054
TOTAL	MAINTENANCE PERSONNEL	221.879	0.505	157.998	142.634	0.290	56.407
	OPERATING PERSONNEL	37.275	0.000	0.000	31.591	0.000	0.000
	HEALTH PHYSICS PERSONNEL	41.702	0.000	39.647	51.373	0.000	24.123
	SUPERVISORY PERSONNEL	17.874	2.000	3.000	9.217	1.261	0.765
	ENGINEERING PERSONNEL	26.399	27.975	26.744	7.561	10.087	10.130
GRAND TOTAL		345.129	30.480	227.389	242.376	11.638	91.425

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information concerning main steam line safety/relief valve challenges for calendar year 1988 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).

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
01/18/88	MS-RV-5B	B	G	E	0%	--
01/18/88	MS-RV-5B	B	G	E	0%	--
01/18/88	MS-RV-3D	B	G	E	0%	--
01/18/88	MS-RV-5C	B	G	E	0%	--
01/18/88	MS-RV-4D	B	G	E	0%	--
01/18/88	MS-RV-4B	B	G	E	0%	--
01/18/88	MS-RV-4A	B	G	E	0%	--

The 01/18/88 actuations were in response to a manual reactor trip following a condenser tube leak forced shutdown.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
02/04/88	MS-RV-1A	B	G	E	0%	88-03
02/04/88	MS-RV-4A	B	G	E	0%	88-03
02/04/88	MS-RV-1B	A	G	A	0%	88-03
02/04/88	MS-RV-2B	B	G	E	0%	88-03
02/04/88	MS-RV-3B	A	G	A	0%	88-03
02/04/88	MS-RV-5B	B	G	E	0%	88-03
02/04/88	MS-RV-1C	A	G	A	0%	88-03
02/04/88	MS-RV-1C	B	G	E	0%	88-03
02/04/88	MS-RV-2C	A	G	A	0%	88-03
02/04/88	MS-RV-2C	B	G	E	0%	88-03
02/04/88	MS-RV-5C	B	G	E	0%	88-03
02/04/88	MS-RV-1D	B	G	E	0%	88-03
02/04/88	MS-RV-3D	B	G	E	0%	88-03

The 02/04/88 actuations were in response to an RPS actuation caused by improper execution of a Technical Specification surveillance procedure.

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

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
04/30/88	MS-RV-5B	C	D	C	0%	--
04/30/88	MS-RV-1A	C	D	C	0%	--
04/30/88	MS-RV-2A	C	D	C	0%	--
04/30/88	MS-RV-3A	C	D	C	0%	--
04/30/88	MS-RV-4A	C	D	C	0%	--

The 04/30/88 actuations were required for setpoint testing.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
06/22/88	MS-RV-1A	B	C	C	9%	--
06/22/88	MS-RV-2A	B	C	C	9%	--
06/22/88	MS-RV-3A	B	C	C	9%	--
06/22/88	MS-RV-4A	B	C	C	9%	--
06/22/88	MS-RV-1B	B	C	C	9%	--
06/22/88	MS-RV-2B	B	C	C	9%	--
06/22/88	MS-RV-3B	B	C	C	9%	--
06/22/88	MS-RV-4B	B	C	C	9%	--
06/22/88	MS-RV-5B	B	C	C	9%	--
06/22/88	MS-RV-1C	B	C	C	9%	--
06/22/88	MS-RV-2C	B	C	C	9%	--
06/22/88	MS-RV-3C	B	C	C	9%	--
06/22/88	MS-RV-4C	B	C	C	9%	--
06/22/88	MS-RV-5C	B	C	C	9%	--
06/22/88	MS-RV-1D	B	C	C	9%	--
06/22/88	MS-RV-2D	B	C	C	9%	--
06/22/88	MS-RV-3D	B	C	C	9%	--
06/22/88	MS-RV-4D	B	C	C	9%	--
06/22/88	MS-RV-3A	B	C	C	9%	--
06/22/88	MS-RV-4A	B	C	C	9%	--

The 06/22/88 actuations were in support of the acoustic monitoring system calibration procedure, Technical Specification requirement 3/4.4.2.

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

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
07/20/88	MS-RV-1B	B	E	C	95%	--
07/20/88	MS-RV-3C	B	E	C	95%	--
07/20/88	MS-RV-2D	B	E	C	95%	--
07/20/88	MS-RV-2A	B	E	C	95%	--
07/20/88	MS-RV-3B	B	E	C	95%	--
07/20/88	MS-RV-3A	B	E	C	95%	--
07/20/88	MS-RV-1C	B	E	C	95%	--

The 07/20/88 actuations were manually cycled to reduce thru seat leakage.

07/21/88	MS-RV-1C	B	E	C	95%	--
07/21/88	MS-RV-1B	B	E	C	95%	--
07/21/88	MS-RV-1B	B	E	C	95%	--

The 07/21/88 actuations were to support post maintenance testing of acoustic monitors.

09/06/88	MS-RV-2C	C	C	C	3%	--
09/06/88	MS-RV-1D	C	C	C	3%	--
09/06/88	MS-RV-5C	C	C	C	3%	--
09/06/88	MS-RV-4D	B	C	C	20%	--
09/06/88	MS-RV-1C	B	C	C	20%	--
09/06/88	MS-RV-3D	B	C	C	20%	--
09/06/88	MS-RV-1B	B	C	C	20%	--

The 09/06/88 actuations were required for setpoint testing and acoustic monitor testing following maintenance.

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

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
09/08/88	MS-RV-4D	B	C	C	15%	--

The 09/08/88 actuation was to support post maintenance testing of an acoustic monitor.

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)	PLANT CONDITION (CODE)	REASON FOR ACTUATION (CODE)	REACTOR POWER LEVEL	ASSOCIATED LER
12/08/88	MS-RV-2A	B	C	C	12%	--
12/08/88	MS-RV-3A	B	C	C	12%	--
12/08/88	MS-RV-1B	B	C	C	12%	--
12/08/88	MS-RV-3B	B	C	C	12%	--
12/08/88	MS-RV-3A	B	C	C	12%	--

The 12/08/88 actuations were in support of post maintenance testing of acoustic monitors.

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 Specifications LCO 3/4.4.2.

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

2.3 SUMMARY OF PLANT OPERATION INCLUDING UNIT SHUTDOWNS/POWER REDUCTIONS

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
1/5/88	S	0	H	5	--	RB	CONROD	Reactor power was reduced, as required, to perform a scheduled control rod sequence exchange.
1/18/88	F	40.8	A	1	--	HC	HTEXCH	The plant was shutdown to correct a condenser tube in-leakage problem. Several leaking tubes were plugged and the reactor was returned to service.
1/20/88	F	6.8	H	1	--	HA	INSTRU	The generator was removed from the grid to recalibrate turbine DEH auto stop oil pressure switches.
2/4/88	F	45.7	G	3	88-03	CD	INSTRU	The reactor automatically shutdown from 100% power due to MSIV isolation caused by improper execution of a Technical Specification surveillance procedure.
2/13/88	F	529.4	A	2	88-06	HC	HTEXCH	The reactor was manually shutdown due to condenser in-leakage and rapidly increasing conductivity. The outage was extended by an inadvertent start of a reactor building supply fan which overpressurized the reactor building, causing the designed roof rupture panels to relieve with resultant damage to the roof.

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

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
3/23/88	F	86.5	A	1	--	HC	HTEXCH	The plant was manually shutdown due to high conductivity as a result of a condenser in-leakage problem.
4/30/88 thru 6/19/88	S	1197.5	C	1	--	RC	FUEL	The plant was shutdown as scheduled for the annual refueling and maintenance outage.
6/24/88	S	1.7	B	1	--	HA	MECFUN	The generator was removed from the grid to perform turbine overspeed testing.
6/24/88	S	77.2	B	1	--	RB	CONROD	The generator was removed from the grid for SCRAM testing. The plant remained shutdown for replacement of a faulty Main Steam Isolation Valve Actuator.
8/19/88	S	0	H	5	--	RB	CONROD	Reactor power was reduced, as required, to perform a scheduled control rod sequence exchange.
8/24/88	F	92.65	D	1	88-29	CI	VALVEX	The plant was shutdown when unidentified leakage in primary containment exceeded Technical Specification limits. A Reactor Core Isolation Cooling (RCIC) valve RCIC-V-63, was identified as the major source of leakage. The valve was repaired and returned to service.

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

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
8/28/88	S	275.85	B	9	--	--	--	The plant remained off line for modifications of Residual Heat Removal check valves, MS-SRV/Vacuum breaker repair and implementation of a design change to the Control Room ventilation system.
9/11/88	F	35.68	A	1	--	SE	--	The generator was removed from the grid and reactor power was reduced to permit drywell entry for inspection of Containment Supply Purge system piping and containment penetration welds.
10/27/88	S	47.97	A	1	--	HB	PIPEXX	The plant was shutdown due to a cracked weld in a main steam line trap station drain. Repairs were made and the plant was returned to service.
12/1/88	F	208.9	A	1	88-037	SA	VALVEX	The plant was shutdown due to air leakage through Containment Supply Purge valves (CSP-V-9) and (CSP-V-5) exceeding Technical Specification limits. Additionally, Main Steam Isolation Valve 28A stuck while opening and required repair prior to plant restart. The valve seats on the CSP valves were replaced and the plant was returned to operation.

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

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE CODE	SHUTDOWN METHOD	LER NUMBER	SYSTEM	COMPONENT	CAUSE AND ACTION TO PREVENT RECURRENCE
12/12/88	S	0	H	5	--	RB	CONROD	Reactor power was reduced to reset control rod pattern for 100% power.
12/30/88	S	0	H	5	--	RB	CONROD	Reactor power was reduced, as required, to perform a scheduled control rod sequence exchange.

CAUSE CODE	TOTAL FOR 1988	TOTAL GENERATOR OFF-LINE HOURS
A	6	949.1
B	3	354.7
C	1	1197.5
D	1	92.6
F	0	0.0
G	1	45.7
H	5	6.8
	TOTAL	2646.4

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

SUMMARY OF CODES

OUTAGE TYPE

F - Forced

S - Scheduled

CAUSE CODE

A - Equipment Failure

B - Maintenance or Test

C - Refueling

D - Regulatory Restriction

E - External Cause

F - Administration

G - Personnel Error

H - Other

SHUTDOWN METHOD

1 - Manual

2 - Manual Scram

3 - Auto Scram

4 - Continued

5 - Reduced Load

9 - Other

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

SYSTEM CODE

STANDARD CODE	SYSTEM DESCRIPTION
CA	Reactor Vessels & Appurtenances
CB	Coolant Recirculation Systems & Controls
CD	Main Steam Isolation Systems & Controls
CF	Residual Heat Removal Systems & Controls
CH	Feedwater Systems & Controls
CI	Reactor Coolant Pressure Boundary Leakage Detection Systems
IA	Reactor Trip Systems
EA	Offsite Power Systems & Controls
EB	AC Onsite Power Systems & Controls
EG	Other Electric Power Systems & Controls
HA	Turbine Generator & Controls
HB	Main Steam Supply Systems & Controls
HC	Main Condenser Systems & Controls
HJ	Other Features of Steam & Power Conversion Systems (not included elsewhere)
MS	Main Steam System

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

SYSTEM CODE

STANDARD CODE	SYSTEM DESCRIPTION
RB	Reactivity Control Systems
RC	Reactor Core
SA	Reactor Containment Systems
SE	Containment Combustible Gas Control Systems & Controls

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

COMPONENT CODE

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Circuit Closers/Interrupters (CKTBRK)	Circuit Breakers Contactors Controllers Starters Switches (other than sensors) Switchgear
Control Rod Drive Mechanism (CONROD)	Control Rod Drive Mechanism
Heat Exchangers (HTEXCH)	Condensers Coolers Evaporators Regenerative Heat Exchangers Steam Generators Fan Coil Units
Instrumentation and Controls (INSTRU)	Controllers Sensors/Detectors/Elements Indicators Differentials Integrators (Totalizers) Power Supplies Recorders Switches Transmitters Computation Modules
Mechanical Function Units (MECFUN)	Mechanical Controllers Governors Gear Boxes Varidrives Couplings

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

COMPONENT CODE

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Penetrations, Primary Containment (PENETR)	Air Locks Personnel Access Fuel Handling Equipment Access Electrical Instrument Line Process Piping
Pipes, Fittings (PIPEXX)	Pipes Fittings
Pumps (PUMPXX)	Pumps
Relays (RELAYXX)	Switchgear
Transformers (TRANSF)	Transformers
Turbines (TURBIN)	Steam Turbines Gas Turbines Hydro Turbines
Valves (VALVEX)	Valves Dampers

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
DG-RLY-DG2/LR	"B" Diesel Generator	During the monthly surveillance, DG2 failed to come up to rated voltage.	During the troubleshooting activities, it was noted that the voltage regulator latching relay was picked up erroneously. This relay was removed and replaced with an approved spare, tested and returned to service.
WMA-TS-12A WMA-TIC-12A2	Control Room - HVAC	During the performance of a surveillance test, temperature switch 12A would not calibrate.	During the trouble shooting evolution, the technicians found that the instrument providing input to the temperature switch required adjustment. No replacement parts were required, the surveillance was completed and the instrument loop was returned to service.
DG-RLY-DG1/K14	"A" Diesel Generator	During the performance of a Technical Specification surveillance, a relay failed and completely deenergized the diesel generator.	The DG #1 control circuit was tagged out and it was noted that the K14 relay had a burned up relay coil. The relay was removed and replaced with an approved spare, tested and returned to service.
FPC-FCV-1 FPC-DPIC-1	Fuel Pool Cooling	The valve/controller for FPC bypass flow is not controlling flow properly.	The flow control valve was isolated and instrument technicians performed an operational recalibration on the associated dP switch. The valve was functionally checked and operated correctly.
HPCS-RLY-5051/A	High Pressure Core Spray	The relay target/seal-in unit will not adjust to the proper value.	The relay was removed and found to be inoperable. The relay was replaced with an approved spare,tested and returned to service.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (Continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
MS-V-28A MS-V-28B MS-V-28C MS-V-28D MS-V-22A MS-V-22B MS-V-22C MS-V-22D	Main Steam	The qualified life of the associated MSIV limit switches and solenoid pilot valves dictates that these components be replaced or rebuilt during the R3 outage.	The limit switches were disconnected and overhauled including the replacement of all consumable components. The associated solenoid pilot valves were removed and replaced with approved spares. The valves were reassembled, functionally tested and returned to service.
RHR-RMS-V/53A	Residual Heat Re-	Valve has "closed" indication regardless of actual valve position.	The position switch contacts were checked for tightness and found to be snug. The position switch was replaced and correct valve position indication restored. The valve was returned to service.
SGT-TS-1B2	Standby Gas Treatment	The high temperature alarm associated with the outlet temperature of HEPA unit B-1 alarms at 80°F rather than 120°F.	The temperature switch was recalibra- functionally tested and returned to service.
CRD-LTS-601A CRD-LTS-601B CRD-LTS-601C CRD-LTS-601D	Control Rod Drive	Scram discharge volume level instruments exceeded trip setpoint - recalibrate.	Recalibrated the scram discharge volume level instruments and returned the loop to service.
DCW-H-1C	Diesel Cooling Water	The immersion heater cycles on/off causing a HPCS system ground.	During the troubleshooting process the immersion heater was found faulted to ground. The heater was replaced, tested and returned to service.

2.4 SIGNIFICANT MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (Continued)

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
MS-V-28A MS-V-28B MS-V-28C MS-V-28D	Main Steam	Main Steam Isolation Valves 28A, 28B, 28C, and 28D failed their local leak rate tests. Disassemble and repair per plant procedures.	The valves were disassembled and overhauled. The repair included machining the valve seats, cleaning the main discs, reassembly and reperforming the local leak rate test. All valves passed the LLRT following overhaul.
HPCS-42-4A7C	High Pressure Core Spray	The breaker for the HPCS keep-filled pump has a broken actuator arm. This breaker cannot be safely energized in this condition.	The defective breaker was removed and replaced with an approved spare, tested and returned to service.
SGT-TC-2B1	Standby Gas Treatment	The temperature switch associated with the discharge of charcoal filter 1B1 would not reset.	The faulty switch was removed, replaced with an approved spare, calibrated and returned to service.
MS-AO-28C MS-AO-28D	Main Steam	General Electric requires a five-year Main Steam Isolation Valve operator overhaul.	Valve operators were removed and replaced with previously rebuilt spares. The replacement operators were installed on the valves and tested satisfactorily.
REA-E/S-613B	Reactor Building Exhaust Air	While performing a Technical Specification test, the power supply for a radiation indicating switch failed.	During troubleshooting it was determined that the power supply for RIS-609B could not be repaired. The power supply was replaced with and approved spare, the surveillance was satisfactorily completed and the instrument loop was returned to service.

2.4 OTHER SIGNIFICANT MAINTENANCE EFFORTS

Main Condenser Inspection and Upgrades

Previous inspections of the Main Steam Condenser had identified problems with condenser tube thinning caused by errosion from the low quality steam dumps and condensate drains to the main condenser. Errosion damage to the tubes had previously caused several unplanned outages. Extensive efforts were taken to install perforated errosion plates and modify high energy baffles to prevent errosion from reoccurring in these areas. Subsequent plant shutdowns and inspections have demonstrated that the modifications were successful in preventing further tube degradation in the modified areas.

Errosion/Corrosion Inspection Program

The errosion/corrosion pipe wall thinning inspection program examined 65 pipe locations during the February and April outages. The effort included a team of seven fulltime NDE Specialists and Engineers. Both water and steam piping from five major systems were examined. Over 20,000 individual wall thickness measurements were made. The trended results indicate that wall thinning is occurring in some of the higher energy steam piping and several drain lines. Pipe wall buildup by welding was completed at four elbows on the Bleed Steam System piping and one tee on the Moisture Separator Reheater drain piping. As a result of data taken during the 1987 outage, the majority of elbows in the eighteen inch Extraction Steam lines were built up on the O.D. by welding. This was done to prevent thru wall errosion prior to the 1989 outage when all carbon steel elbows in this pipe run will be replaced with stainless steel elbows. All of the thickness measurements taken have been input into the data base for trending and remaining life predictions.

Heat Exchanger Tube Integrity Program

The heat exchanger tube integrity program examined over 12,000 heat exchanger tubes or the equivalent of 65 miles of tubing. Eddy current inspection identified leaking and flawed tubes in Feedwater Heaters 6A and 6B, Main Steam Condenser and the Reactor Closed Cooling Heat Exchangers A, B, and C. Based on the data, criteria for preventive plugging was developed. In addition, several tubes were removed and metallurgical analysis was performed on the defects. Samples of the tubes were analyzed using an electron microscope. Stress corrosion cracking was identified in the Feedwater Heater 6A and 6B tubing. Based on the inspection date and future trending, accurate estimates of remaining useful life can be made. This effort involved four fulltime Supply System Engineers and NDE Specialists, and twelve contract employees.

Westinghouse 480 Volt Circuit Breaker Inspection

In response to NRC Bulletin 88-01, "Defects in Westinghouse Circuit Breakers", the Supply System performed short-term and long-term inspections on twenty-one (21) type DS-416 480 volt A.C. circuit breakers in Class 1E safety related applications. Six (6) of the 21 circuit breakers passed the required acceptance criteria for both the short-term and the long-term inspections with no limitations. Three (3) of the remaining 15 circuit breakers failed to pass the long-term required acceptance criteria based upon improper roller alignment. Spare breakers were located to replace the three failed breakers. One of these three spare breakers also failed the roller alignment criteria, an additional spare breaker was located and installed. Fifteen (15) circuit breakers (including the replacement breakers) passed the weld inspection criteria for limited use, which will require periodic reinspections, or replacement of breaker pole shafts. The long-term inspection was successfully performed on these breakers.

"B" Loop Hydraulic Power Unit and Flow Control Valve Actuator Refurbishment

The "B" Reactor Recirculation Pump Hydraulic Power Unit (HPU) and the Flow Control Valve (FCV) Actuator were disassembled and refurbished during the R-3 outage as part of a preventative maintenance program designed to reduce nuisance leakage from the HPUs. The refurbishment included the cleaning and polishing of the inside of the cylinder, the installation of new seals, and tolerance verification. The flow control valve actuator was sent to the manufacturer for a complete overhaul. By implementing an extensive maintenance program on these components, the potential for introduction of fyrquel (hydraulic fluid), into the radwaste system is reduced.

One-Year Diesel Engine Maintenance

The Supply System performed the manufacturers' recommended one-year preventative maintenance to all five of the WNP-2 diesel engines. This maintenance included, but was not limited to, replacement of all filter elements in the lube oil, fuel oil and intake air systems, engine one-revolution inspection, and inspection of all of the turbo charger after coolers. Additionally, the power pack assemblies on the 1A2 and 1B2 engines were replaced. All equipment was found to be free of excessive wear or degradation.

Main Steam Bypass Valve Inspection

During the 1987 refueling/maintenance (R2) outage, all four main steam bypass valves were modified to reduce steam loss. As a followup action to that modification, a special test was performed during the 1988 outage to determine the Balance of Plant (BOP) steam loss with all turbine bleed steam isolation valves closed. The conclusion, following this test, was that the total BOP main steam loss was approximately equivalent to a 1/2" steam line, routed to the main condenser. In an attempt to verify this test conclusion, two of the modified valves were disassembled and inspected during the R3 outage. The valves had approximately zero leakage following one year of service. During the 1989 maintenance outage, the remaining two valves will be disassembled and inspected.

Turbine Maintenance

During the 1988 refueling outage, the High Pressure turbine was disassembled for manufacturer's required inspections. Visual inspections were performed with only minor errosion found at the horizontal joint of the HP shell at both blade ring fits and at the horizontal joint of the inner glands. No unacceptable indication of any cracks were found. Additionally, visual inspections were performed on the reheat valves, the turbine oil reservoir, the HP turbine discharge piping and the last stage blades of the Low Pressure turbine. No unacceptable indications of cracking, errosion or any other abnormality were found during these inspections.

Main Turbine Throttle Valves

Prior to plant shutdown, a test was performed to determine if WNP-2 could repeat a scenario from a Westinghouse Advisory Letter 87-03, where the throttle valves "stick open". During the performance of this test, we were able to get one out of four valves to "stick open". An in-house root cause evaluation was performed and a minor modification was made to the valves which entailed changing some insulation on the throttle valve bonnet at the cross head support location and increasing the clearance from the cross head to the cross head support to the maximum allowable. Following these modifications, another test was performed, and the "sticking" condition was not repeated.

Main Turbine Governor Valves

During the 1988 refueling outage, WNP-2 implemented a manufacturers' recommended "stiff stem and bolted bushing" modification to the #1 and #4 governor valves. This modification was implemented in attempt to solve the governor valve vibration problem which has been occurring at WNP-2 for 2 years. During the valve disassembly, the valve configuration was modified by adding a relief groove in the plug seal ring groove to eliminate a theorized valve binding scenario. During the 1989 refueling outage, an inspection will be made to evaluate the effectiveness of this modification. Additionally, a complete dimensional mapping was performed on all four governor valves to evaluate the current valve configuration and for future historical data.

2.4 OTHER SIGNIFICANT MAINTENANCE EFFORTS (Continued)

Implementation of the MOVATS Program

The MOVATS testing program for the 1988 refueling outage included twenty-one (21) motor operators (MO) which had not been previously tested and seven (7) post maintenance tests of MO's which had been tested during the 1987 outage. All seven of the followup tests were completed with satisfactory test results. During the testing, a generic problem with torque switches in SMB-000 MO's was discovered. The problem switches are made of melamine and exhibit two failure modes; cam binding and broken cam lugs. All torque switches for safety related SMB-000 MO's made of melamine were replaced with torque switches of a different material and with metal cam lugs. In addition, a sample of torque switches of other materials was inspected to assure the problem was limited to melamine.

The scope of torque switch replacements was expanded when Equipment Qualification Engineering determined that a family of SMB-00 MO's had torque switches of similar design and material as those in SMB-000 MO's. All such torque switches were replaced or justified as not needing replacement. The torque switch failures were determined to be reportable per 10CFR Part 21 and reported via LER 88-17.

2.5 INDICATIONS OF FAILED FUEL

INTRODUCTION

In accordance with the commitment and requirements described in WNP-2 FSAR, Section 4.2.4.3, a visual inspection of discharged fuel assemblies from WNP-2, Cycle 3 was performed on October 10-11, 1988. The purpose of the inspection was to verify assembly and rod structural integrity. In addition, a visual inspection of discharged fuel channels was performed at the same time.

SUMMARY OF INSPECTION RESULTS

A total of nine assemblies and four channels discharged at the end of Cycle 3 were inspected. No evidence of geometric distortion, rod bow or cladding defects were observed. The fuel did exhibit nodular corrosion which covered portions of the cladding on some of the fuel rods inspected. The extent of coverage did not appear to have changed significantly when compared to the fuel discharged at the end of Cycle 2. Therefore, it is concluded that the rate of nodular growth appears to have slowed. This level of crud induced localized nodular corrosion coverage is most closely related to G.E. visual standard No. 2.

A foreign object was observed resting on the lower tie plate between two fuel rods in one assembly. During Cycle 3, a fuel failure believed to be a single rod leaker was detected about mid-way through the cycle. Flux testing conducted during the cycle identified several suspected assemblies. The reduction in off-gas activity noted during Cycle 4 indicates the leaking assembly was discharged (refer to Section 2.9 of this report).

Fretting marks were noted on several assemblies, particularly in the region of the sixth grid. The marks appear to be a slightly polished area on the clad and may have been caused during the dechanneling operation.

The inspected channels all exhibit a generous coating of a flake-like oxide layer. Some scratches were noted and two possible pre-hole locations were observed.

SELECTION OF ASSEMBLIES AND CHANNELS

During the spring 1988 refueling outage, 152 assemblies were discharged. Nine of these assemblies and four channels were selected for visual inspection. The nine assemblies represent greater than 5 percent of the discharged fuel and are representative of the highest burnup assemblies in the discharged batch. The visual examination of the peripheral rods included observations for cladding defects, fretting, rod bow, missing components, corrosion, crud disposition and geometric distortions. The four channels selected had experienced weld repair during manufacturing.

Of the 152 assemblies discharged, there are 38 sets of four assemblies with approximately the same exposure and power history. Assemblies inspected were selected to provide a range of both exposure and power history. In addition, preference was given to assemblies which underwent power increases greater than 10 percent in Cycle 3. The selected assemblies are all high enriched (2.19 weight percent U-235) assemblies. Some characteristics of the selected assemblies are shown in Table 1.

TABLE 1.0
CYCLE 3 DISCHARGED FUEL ASSEMBLIES SELECTED FOR EXAMINATION

FUEL ASSEMBLY IDENTIFICATION	EXPOSURE (MWD/MTU)	CYCLE 3 POWER INCREASE	CHANNEL FEATURES	COMMENT
LJT 465	18,322	- 37.3% (Decrease)	Weld Repair Channel	
LJT 424	20,703	16.1%		Next to Suspect Leaker Cell
LJT 646	20,462	- 15.0%		1/8 Core Symmetric to LJT 424
LJT 814	20,644	14.9%	Weld Repair Channel	
LJT 417	21,588	- 13.3%	Weld Repair Channel	
LJT 658	20,121	- 19.9%	Weld Repair Channel	From Suspect Leaker Cell; Highest Power In Cycle
LJT 719	18,944	- 17.0%		From Core Center
LJT 709	20,614	15.1%		1/8 Core Symmetic to LJT 814
LJT 444	20,570	0.0%		

The nine assemblies inspected have exposures from 18,322 to 21,588 MWD/MTU. The highest (110% core average) and lowest (58% core average) power asseblies for Cycle 3 are included, and represent power changes from -37% to +16% from those experienced in Cycle 2. An assembly from a suspect leaker cell and an assembly adjacent to a suspect leaker cell were included in the inspection.

INSPECTION TECHNIQUE

The poolside visual examination was performed with an underwater periscope system and the results of the fuel inspection recorded with a 35mm camera. In general, two sides of each fuel assembly were viewed. Photographs were taken of the points of interest. A total of 60 photographs of the examined fuel and channels were taken. The inspection procedure involved moving the selected fuel assembly in a vertical position past the fixed periscope. This was accomplished by raising the fuel assembly out of the spent fuel rack with the fuel handling mast on the refuel bridge. Channel inspection was performed in a similar manner. An assembly wash station was used to remove surface crud from some of the edge fuel pins in order to assess the rate of nodular growth under the surface corrosion (crud).

INSPECTION CRITERIA

Visual inspection of the selected assemblies was performed to determine the extent of the following phenomena:

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

RESULTS OF THE FUEL EXAMINATION

The inspected fuel assemblies appeared to have good structural integrity. The upper tie plates were level, fuel pin springs had ample compression space, tie rod nuts were snug and all of the fuel pins observed were properly seated in the lower tie plate. The spacers appeared perpendicular to the fuel pins and were properly located. No finger spring damage was observed. The grid spacers in general exhibit a heavy nodular buildup which appears to have saturated based on comparsion with past observations. The only anomoly was the observation of a foreign object observed on the lower tie plate of LJT 658. This object appeared to be of sufficient size that entry from either the lower or upper tie plate is improbable.

2.6 PLANT MODIFICATIONS

Federal Regulations (10CFR50.59) and the Facility Operating License (NPF-21) allow changes to be made to the facility 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 1988 are provided. Included are summaries of the safety evaluations.

2.6.1 PLANT DESIGN CHANGES

The following plant design changes were completed in 1988 and reported in accordance with 10CFR50.59. These modifications were evaluated and it was determined that they did not (a) increase the probability of occurrence of an accident or malfunction of the equipment important to safety, as previously evaluated in the WNP-2 updated Final Safety Analysis Report (FSAR), (b) create the possibility of an accident or malfunction of a different type than previously evaluated in the FSAR, (c) reduce the margin of safety as defined in the basis for any WNP-2 Technical Specifications, or (d) require a change to the WNP-2 Technical Specifications.

2.6.1 PLANT DESIGN CHANGES (Continued)

PLANT DESIGN CHANGE 84-1394

Plant Design Change 84-1394 was initiated to replace the originally installed Reactor Water Cleanup, (RWCU) pumps which had a history of frequent failures. High maintenance frequencies and long replacement part lead times from the vendor required the Supply System to maintain a costly large spare parts inventory. Additionally, pump overhaul attributed to radiation exposure to maintenance personnel ranging from 1.5 to 3.5 Man-Rem per pump overhaul.

This modification installed two Hayward Tyler sealess pumps. Each pump has a rating of 480 GPM with a total system flow capability of 133 percent. The pumps are expected to increase system reliability, reduce maintenance and improve reactor water quality with the systems additional flow capacity.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the system capability or function was not reduced in any manner. The additional capacity has not been used pending completion of an engineering stress analysis.

PLANT DESIGN CHANGES 85-0115-1 & 86-0624

Plant Design Changes 85-0115-1 and 86-0624-0 were initiated to provide additional isolation versatility to subsystems of the fire protection system. This modification provides increased versatility to isolate portions of the fire protection system as a result of damage, surveillance, or maintenance, thus minimizing the decrease in fire protection coverage. This increases the system availability and reliability.

These modifications installed manual isolation valves on fire protection lines to provide additional isolation versatility. Also, a sleeve was installed on a line running under the diesel generator building, reducing the probability of damage to this line from a safe shutdown earthquake.

These modifications did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the increased availability and reliability of the fire protection system derived from increased isolation versatility outweighs the negligible decrease in system reliability caused by increasing the number of valves. This is because the valves added are manually repositioned, locked, and position verification performed monthly, which results in an acceptably low probability of failure.

2.6.1 PLANT DESIGN CHANGES (Continued)

PLANT DESIGN CHANGE 85-0115

Plant Design Change 85-0115 was initiated to install automatic sprinklers for fire protection in a high fire loading area in the Radwaste Building. Also, this satisfies a related recommendation from the (ANI Recommendation 84-11a) and accounts for a modification in the fire loading for the area protected. The area is now used for temporary storage of transitory combustibles such as those accumulated during a refueling outage.

This modification installed automatic sprinklers and the associated electrical alarm wiring for the truck bay and storage area in the 437 foot elevation of the Radwaste Building.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) this installation is not a safety related system; (2) an evaluation determined that water in this area would not affect the performance of any safety related equipment; and (3) fire protection in this area reduces the probability of damage to safety related equipment and/or unintentional release of radionuclides to the environment caused by a fire in this area.

PLANT DESIGN CHANGE 85-0131

Plant Design Change 85-0131 was initiated to prevent debris from entering the weir box in the Circulating Water Pump House (CWPH) and becoming entrained in the Circulating Water and Plant Service Water (TSW) Systems. This reduces fouling in downstream components which increases the reliability of these components served by the TSW and decreases the probability of unplanned transient events.

This modification lowered the CWPH weir box vent stacks and installed steel "Q" decking over the CWPH basin.

This modification did not involve a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the circulating water, tower makeup, and plant service water systems are not safety related. Also, damage or malfunction of any of these systems would not affect the performance of any safety related equipment.

PLANT DESIGN CHANGE 85-0545

Plant Design Change 85-0545 was initiated to provide standby AC power to the Cooling Jacket Water (CJW) pumps that supply cooling water to the plant control air compressors. Standby AC power was already available to the compressors prior to the modification. The availability of the CJW pumps allows for continued operation of the compressors following a Loss of Offsite Power event. The continued availability of the control air compressors would provide air supply to non-safety related components and instruments. This results in making more components available to mitigate the event, therby minimizing the consequences of the event and increasing the probability of a successful shutdown.

This modification connected the operating and backup CJW pumps to Division 1 and Division 2 emergency power sources. Only one pump is needed to circulate cooling water to all three air compressors. Auto-standby controls were provided to automatically start the backup pump in the event the operating pump failed or lost power.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the increased load from the CJW pumps was already included in the FSAR evaluation, and (2) the continued availability of the air compressors following a loss of offsite power makes more components available to minimize the consequences of the event, increasing the probability of a successful shutdown.

PLANT DESIGN CHANGE 85-0564

Plant Design Change 85-0564 was initiated because the auto advance function of the intake air filters for the Control Room, Radwaste, Turbine and Reactor Buildings did not operate properly with the originally installed dP switches. The wide variations in air flow caused the dP switched to actuate the auto filter advance process unnecessarily, thereby wasting roll filters.

This design change replaced the originally installed dP switches with auto advance timers for all roll type intake air filters located in the Control Room, Radwaste, Turbine and Reactor Buildings.

The installation of the automatic advance timers did not require a modification to the WNP-2 Technical Specifications or involve an unreviewed safety question because the new timers increase the reliability of the system and have not affected the functional characteristics of the system.

PLANT DESIGN CHANGE 85-0718

Plant Design Change 85-0718 was initiated to resolve a gap in gaseous effluent monitoring between the normal and extended range monitors. Due to a Off Site Dose Calculation Manual requirement, the Hi-Hi setpoint was set at 2 decades less than the top of scale on the normal sample rack. By design, the Hi-Hi trip signal transfers the flow to the extended range monitor, which incorporates a one decade overlap with the normal range, resulting in a monitoring gap of approximately 1 decade.

Rather than revise the ODCM calculation and bases for the Hi-Hi setpoint, a design modification to route the sample through both sample racks continuously rather than the previous transfer arrangement was implemented. This design provides continuous monitoring capabilities throughout all monitoring ranges.

There were no modifications to the WNP-2 Technical Specifications as a result of this design change. This change did not involve an unreviewed safety question because with the continuous monitoring capabilities, the system is more reliable and eliminates the possibility of any sample gaps.

PLANT DESIGN CHANGE 86-0229

Plant Design Change 86-0229 was initiated in response to requirements set forth in 10CFR50.62 "Requirements for Reduction of Risk from Anticipated Transients without SCRAM (ATWS) Events for Light-Water-Cooled Nuclear Power Plants".

An Alternate Rod Insertion (ARI) system was installed at WNP-2 during the refueling/maintenance outage of 1988. This system was designed to insert all control rods in the event of an ATWS condition (Low Level 2 or High Rx pressure without a SCRAM). The installed system consists of manual controls and a logic system in the main control room, eight new instruments installed in two plant instrument racks and nine new solenoid valves installed in strategic locations designed to vent the Control Rod Drive air header which in turn will open the individual scram inlet and outlet valves, thereby automatically inserting the control rods.

This plant modification did not result in a modification to the WNP-2 Technical Specifications (as approved by the NRC endorsement of the BWROG proposal) or result in an unreviewed safety question because the ARI system is designed to actuate in the event the RPS trip system does not scram the plant. Therefore, as a backup system to RPS, the ATWS-ARI system increases the margin of a safe shutdown.

2.6.1 PLANT DESIGN CHANGES (Continued)

PLANT DESIGN CHANGE 86-0343

Plant Design Change 86-0343 was initiated as a result of a fire protection walkdown which noted that the Shift Manager's office window was not constructed with a fire retardant material and therefore was not bounded by the combustible material loading evaluation.

This modification replaced the originally installed window with a one-hour fire rated window and in addition added fire retardant wall paneling, book shelves, office furniture and carpet. An aluminum suspended ceiling and a fire stop at the south wall line between the existing computer floor and concrete sub-floor was also installed.

This modification did not result to a change in the WNP-2 Technical Specifications or involve an unreviewed safety question because the reduction of combustible material in the control room reduces the consequences of damage to safety related equipment.

PLANT DESIGN CHANGE 86-0618

Plant Design Change 86-0618 was initiated to provide improved control during the shutdown cooling mode. The result of the modification is reduced wear on the Residual Heat Removal (RHR) system components resulting in improved RHR System reliability.

This modification installed control circuitry to allow throttling of the RHR heat exchanger discharge valves (RHR-V-3A&B) during the shutdown cooling mode. The RHR-V-3A&B valves and the RHR heat exchanger bypass valves (RHR-V-48A&B) are used to control cooling during the shutdown cooling mode. The previous method of control used the RHR-V-48A&B and the cooldown injection valves (RHR-V-53A&B), which resulted in significant wear in the RHR-V-53A&B valves.

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

PLANT DESIGN CHANGE 87-0009

Plant Design Change 87-0009 was initiated because the originally designed and installed Traversing In-Core Probe (TIP) System containment isolation valves did not meet the specified ASME design criteria. The configuration remained unchanged as presented in the FSAR.

This modification removed the originally installed ball valves, containment penetration flanges and the tubing between the outboard valves and the flanges, and replaced them with ASME qualified components. The design or function of these components did not change as a result of this modification. The design change was implemented to ensure verbatim compliance with design criteria per 10CFR50 Appendix A.

This modification did not result in a change to the WNP-2 Technical Specification or result in an unreviewed safety question because: (1) the margin of safety was not reduced by upgrading the components and (2) the functional design of the system was not altered as a result of this design change.

PLANT DESIGN CHANGE 87-0085

Plant Design Change 87-0085 was initiated to ensure control room indication from the Circulating Water Pump House (CWPH) and circulating water equipment control would be available following a loss-of-offsite power or a LOCA event. This results in making information and components available to mitigate the event, thereby minimizing the consequences of the event and increasing the probability of a successful shutdown.

This modification provides standby power to the supervisory panels in the CWPH from the Division 1 and 2 diesel generators (DG).

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the increased load to the DGs from the supervisory panels does not exceed the allowable DG continuous load rating; (2) the increased load to the DGs was included in the FSAR evaluation; and (3) the continued availability of control room indication from the CWPH and equipment control makes more information and components available to minimize the consequences of the event, increasing the probability of a successful shutdown.

PLANT DESIGN CHANGE 87-0160

Plant Design Change 87-0160 was initiated as a result of concerns noted by a Nuclear Regulatory Commission Inspector during a routine plant tour. The major concerns were that personnel were not always notifying Health Physics to perform a radiation survey prior to working on the refueling bridge, and that all equipment being removed from the Fuel Pool was not being immediately surveyed upon removal as required by approved plant procedures.

This plant modification installed a dedicated power supply and mounting bracket on the refuel bridge platform to accommodate the installation of an Area Radiation Monitor (ARM). Prior to the start of refuel activities, the ARM is installed on the refuel bridge by the Health Physics organization as required by plant procedures.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the addition of the ARM has increased personnel safety and enhances the radiation monitoring capabilities at WNP-2.

PLANT DESIGN CHANGE 87-0229

Plant Design Change 87-0229 was initiated to provide improved air filtration to the Air Inlet Room of the High Pressure Core Spray (HPCS) Emergency Diesel Generator (DG). This modification provides cleaner intake air to the diesel engine, particularly when the engine is running at idle speeds and during periods of high ambient dust concentrations and volcanic ashfall. The existing oil bath filter for the diesel engine, which is located in the Air Inlet Room, is inefficient at low engine speeds. As result, the modification reduces engine wear per unit time of operation, thereby increasing engine reliability.

This design change installed air filters at the inlet of the Air Inlet Room, upstream of the diesel engine oil bath filter, and removed the HVAC prefilters on the outlet from the Air Inlet Room to the HVAC. Also, a delta-pressure gage was installed across the new filters to provide filter loading indication. To ensure filter integrity and limit air flow resistance, the filters are replaced at or prior to reaching a maximum allowable delta-pressure.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the DG reliability is increased by reduced engine wear through improved intake air filtration. In the unlikely event filter replacement is severely neglected and the filters become heavily loaded, the filters are designed to fail prior to any loss of engine power caused by restricted air flow allowing an unrestricted flow path to the engine air intake lines through the existing oil bath filter.

PLANT DESIGN CHANGE 87-0328

Plant Design Change 87-0328 was initiated to improve the reliability of the drains to the diesel engine starting air dryers, and to reduce the frequency and magnitude of repair to the dryer drains. The improved reliability of the drain system reduces the probability of fouling downstream components which results in increased reliability of the diesel generators.

The automatic drain traps were removed by this design change, and the dryer drain lines were replaced with stainless steel parts and piping to reduce corrosion to drain piping. Manual blowdown of the drains is performed periodically.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the reliability of the air dryer drain system was improved, thereby increasing the reliability of the diesel generators.

PLANT DESIGN CHANGE 88-0026

Plant Design Change 88-0026 was initiated to provide additional containment isolation capability for the Reactor Core Isolation Cooling (RCIC) system. This reduces the probability of the RCIC suction line becoming a containment leakage path to the environment and satisfies the requirements for containment isolation and leakage mitigation criteria as specified by the NRC. (For further information see LER 88-002).

This modification installed a new check valve (RCIC-V-204) downstream of the motor operated Condensate Storage Tank (CST) valve (RCIC-V-10) and relocated the 2-inch RCIC keep full pump suction line (2" RCIC-(8)-1-1) downstream of RCIC-V-204. In the event of a design basis LOCA concurrent with a seismic event and a single component failure, the two-inch RCIC keep full pump suction line would become a leakage path from the suppression pool to the environment without the check valve.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the low probability of a check valve to fail closed negligibly reduces the reliability of the RCIC system, (2) increases the probability of successful containment isolation following an event requiring isolation, (3) the boundary conditions used in the FSAR were not affected, and (4) the WNP-2 Technical Specifications margins were not altered.

PLANT DESIGN CHANGE 88-0060

Plant Design Change 88-0060 was initiated to replace the built-up roofing and insulation on the reactor building that was damaged during the Reactor Building overpressurization event on February 14, 1988.

This modification replaced the damaged reactor building built-up roof with an elastic sheet membrane roof system. This roof system is designed to allow over-pressure relief prior to any structural damage being done to the reactor building structure during a design basis tornado, maximum credible overpressurization event.

The modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the reactor building roof membrane replacement does not affect the boundary conditions used in the FSAR evaluations or affect the performance of safety related equipment during FSAR analyzed accident conditions.

PLANT DESIGN CHANGE 88-0079

Plant Design Change 88-0079 was initiated to replace two pressure switches that provide input to the Main Steam Safety Relief Valve Actuators. These pressure switches had degraded such that they needed to be replaced, however, the manufacturer no longer supplies Class I switches and no spare parts were available. This design change replaced the originally installed pressure switches with new switches that met the Class I design specifications.

This modification did not result in a change to the WNP-2 Technical Specifications or result in an unreviewed safety question because the replacement switches perform the same function as the originally installed equipment, they meet the required Class I design specifications and the devices are not covered in the WNP-2 Technical Specifications.

2.6.1 PLANT DESIGN CHANGES (Continued)

PLANT DESIGN CHANGE 88-0151

Plant Design Change 88-0151 was initiated to ensure the Residual Heat Removal System (RHR) suppression pool suction valves would not be opened prior to closing the RHR shutdown cooling suction valves. The modification prevents a valve lineup that would unintentionally drain the reactor vessel to the suppression pool.

This modification changes the open control logic for the suppression pool suction valves (RHR-V-4A&B) by adding an interlock that prevents RHR-V-4A&B from opening if the shutdown cooling valves (RHR-V-6A&B) are open. This modification also removes the seal-in from the opening control circuit of the RHR-V-4A&B to allow manual control at any time. Previously, the valve open circuit would seal-in and the valve could not be closed until it reached the full open position.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the changes significantly reduced the probability of an inadvertent lineup in the Residual Heat Removal (RHR) system that would drain the reactor vessel, and (2) the boundary conditions of the FSAR evaluations remained unchanged.

PLANT DESIGN CHANGE 88-0188

Plant Design Change 88-0188 was initiated to relocate selected Rod Position Indication (RPI) cables to spare penetration circuits. This modification restored RPI operability to two control rods without changing the safety function of the penetrations.

This plant design change relocates control rod 46-07 and 10-39 RPI system cables to spare penetration circuits. The RPI circuits for those rods were found to have open circuit wires in their penetrations.

This modification did not result in a change to WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the safety function of the penetration was not changed, and (2) the boundary condition for the FSAR evaluation was not changed.

2.6.2 LIFTED LEADS AND JUMPERS

The following are summaries of noteworthy modifications made to the plant by the use of lifted leads and jumpers during 1988. Each modification was evaluated and determined not to represent an unreviewed safety question or require a change to the WNP-2 Technical Specifications.

Electrical Jumper on Refueling Bridge Cable Logic Leads

An electrical jumper was installed around the broken control logic leads in the refueling bridge cable that rendered the "over-the-core" interlock inoperable. The broken leads resulted in a rod block, thus preventing start-up. The modification returned to service the Reactor Manual Control System (RMCS) that allowed withdrawal of the control rods for reactor operation. At the conclusion of R3, when parts were available, the broken cable was replaced.

The installation of this jumper did not involve an unreviewed safety question or reduce any margin of safety as defined in the WNP-2 Technical Specifications. With the reactor fully assembled, and the drywell head and shield blocks installed, the interlocks are not required to be functional. The Technical Specifications require the refueling bridge interlocks be operable only when in the refueling mode. Since the plant was not in the refueling mode nor relying upon operation of the refueling bridge when the jumper was applied, the margin of safety was not affected.

Lifted Leads on a Main Steam Line Drain Valve Inoperable Status Indication

A Bypass and Inoperable Status Indication (BISI) relay was jumpered for the deenergized main steam line drain valve, MS-V-67A, to clear the out-of-service alarm, this action allowed possible additional alarms to be recognized. The deenergization of the valve causes a BISI alarm, masking any other out-of-service component alarms. The valve was deenergized and in the shut position.

This relay removal and jumper did not involve an unreviewed safety question or reduce the margin of safety as defined in the WNP-2 Technical Specifications because: (1) the valve was deenergized in the closed position, capable of performing the FSAR analyzed safety function without changing state; (2) the shut valve causes a negligible affect on the reliability of normal operating systems resulting in a negligible change in probability of an unplanned transient event; and (3) the lifted relay removes the known and accepted alarm condition, and unmasks potential BISI alarms from other components, reestablishing original the level of status information to the control room.

Lifted Leads on a RHR Shutdown Cooling Return Line Valve Inoperable Status Indication

A Bypass and Inoperable Status Indication (BISI) relay was lifted for a deenergized cooldown injection valve, RHR-V-53A, to clear the out-of-service alarm, allowing possible additional alarms to be recognized. The deenergization of the valve causes a BISI alarm, masking any other out-of-service alarms. The valve was deenergized and in the shut position.

This lifted relay did not involve an unreviewed safety question or reduce the margin of safety as defined in the WNP-2 Technical Specifications because: (1) the valve was deenergized in the closed position, capable of performing the FSAR analyzed safety functions without changing state; (2) the shut valve has no affect on the reliability of normal operating systems resulting in no change in the probability of an unplanned transient event; and (3) the lifted relay removes the known and accepted alarm condition, and unmasks potential BISI alarms from other components, reestablishing the necessary level of status information to the control room.

2.6.3 FSAR AMENDMENT EVALUATIONS

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

Chapter 6, Containment Sprays

Modification - This revision to the FSAR deletes the requirement for a 15 second maximum closure time for the Residual Heat Removal (RHR) system minimum flow valves.

Basis for Change - This change makes the FSAR consistent with the WNP-2 Technical Specifications and allows the two year Valve Position Indication (VPI) program requirements to be satisfied. In order to satisfy the VPI requirements for stroke and stroke rate, the open limit switch had to be adjusted for further opening of the valve. As a result, the FSAR 15 second maximum closure time could not be met. However, the valve stroke time is not required per the Technical Specifications since the valve does not receive an isolation signal. Also, an engineering evaluation determined that 1) the valves travel at the same rate and are within the travel rate design guidelines (four inches per minute), 2) the valves will be open to the same position at the same time as measured during Plant Startup Testing, and 3) the further opening will have a negligible effect because the largest reduction in flow/pressure is caused by three restriction orifices downstream. Consequently, this modification does not constitute a facility change since component function and purpose are not altered or limited, and component or system reliability is not reduced.

Chapter 6, Containment Sprays

Modification - This modification lowers the Low Pressure Core Spray (LPCS) relief valve setpoints on the LPCS-RV-18 and LPCS-RV-31 from 550 psig and 100 psig to 427 psig and 97 psig, respectively.

Basis for Change - This modification was made to ensure adequate overpressure protection on the LPCS pump (LPCS-P-1) suction and discharge lines. This modification reflects setpoints calculated in the analysis documentation and does not involve an reviewed safety question because the margin to overpressure protection of the LPCS piping is increased.

Appendix B, WNP-2 Response to Regulatory Issues Resulting from TMI-2

Modification - This revision changes: (1) the liquid sample dilutant from 100 ml to 10 ml (Page B.2-16a); (2) the sample bottle size from 27 ml to 21 ml (Page B.2.16b); and (3) the quarterly operability testing, analysis comparison, personnel training, and annual drill to reflect the current practices (B.2-16c).

Basis for Change - This modification was made to correct sample volume errors and to clarify quarterly operability testing, sample analysis comparison, and training and drill criteria to reflect the actual practices that meet the requirements of TMI-2 Section II.B.3 for post-accident sampling station samples.

Chapter 7, Safety Related Display Information

Modification - This change adds descriptions of post-accident monitors consistent with the as-built plant conditions and corrects inaccuracies.

Basis for Change - This change updates the FSAR to accurately reflect WNP-2 completed commitment to R.G 1.97 R2. An unreviewed safety question was not involved because no revisions were made to the existing system operations and the changes had no effect on the Technical Specification margins.

Chapter 7, Main Steam Line-Tunnel High Differential Temperature

Modification - Changes were made to clarify the actual plant configuration and operation of the Main Steam-Line Tunnel high differential temperature channels.

Basis for Change - The changes more accurately describe the actual plant configuration. An unresolved safety question was not involved because: (1) the revised description of the plant configuration remains within the FSAR boundary conditions and evaluations, and (2) the revisions had no effect on the Technical Specification margins.

Chapter 11, Solid Waste Management System

Modification - The radwaste processing contractor was changed to another vendor. The section was revised to reflect current operations using Pacific Nuclear for radwaste processing.

Basis for Change - Changing the radwaste contractor will not significantly alter the processes or methods employed to safely process the radwaste. The same activities, types and quantities of radwaste are involved as before the contractor change. This change does not result in a change to the WNP-2 Technical Specification margins and does not involve an unreviewed safety question because: (1) the processes performed are not significantly different, (2) the radwaste will continue to be handled safely in accordance with 10CFR50.61, and (3) the boundary conditions for the FSAR evaluations and the WNP-2 Technical Specification margins were not changed.

2.6.4 OTHER

Included in the Plant Nonconformance Reporting (NCR) process at WNP-2 is the requirement to perform a 10CFR50.59 Evaluation for those NCRs which are dispositioned as "Use-As-Is," "Repair," or "Conditional Release." The specific purpose of the 10CFR50.59 evaluation is to recognize these categories of NCRs as implementing a change to the facility, thus requiring a 10CFR50.59 evaluation. When a 10CFR50.59 Safety Evaluation is performed, the NCR is reviewed by the Plant Operating Committee and approved by the Plant Manager prior to the equipment being declared operable.

The following is a discussion of plant changes which were made by means of the NCR process during 1988:

NCR 288-050 (Reactor Building Roof Rupture Due to HVAC Overpressurization Transient)

- Problem Description

  Rupture of the reactor building roof occurred as a result of a HVAC overpressurization transient. Overpressurization resulted from operation of a return outside air fan without associated exhaust air fan in operation due to logic circuit wiring error. The reactor building roof (secondary containment) was restored to as-built conditions prior to operation, except for installation of the environmental covering.

- Corrective Action

  The immediate disposition ("Repair" and "Use-As-Is") included: (1) analysis of the release fasteners which concluded the roof ruptured as designed, (2) additional compensatory measures to prevent another roof rupture, and (3) restoration of the roof to satisfy the FSAR defined protective and secondary containment functions. The environmental covering was not installed on the roof prior to operation of WNP-2 because it does not provide a protective or secondary containment function and, as a result, does not affect secondary containment integrity.

  The protective function of the roof of the reactor building is to contain a High Energy Line Break (HELB) per FSAR Section 6.2.3 (limit is 0.25 psid) and rupture at 0.5 psid per the high wind and tornado design bases (FSAR Section 3.3.2). An engineering evaluation of the restored roof configuration determined the roof would rupture prior to the 0.5 psid, which is conservative and acceptable, and would contain the 0.25 psid HELB. The strength of the environmental covering was conservatively not considered in the failure mechanism of the roof, but the weight of the covering is considered in the loading of the roof members. The roofing material does not provide missile protection (FSAR Section 3.5).

  A building leak test without the environmental covering satisfied the Technical Specification leakage criteria, i.e., less than 2240 CFM at 0.25 inches w.c. A single Standby Gas Treatment (SGT) train is verified to satisfy the minimum allowable capacity requirements every 18 months.

The missing roof covering was considered only as an environmental barrier for the structural material. Any affects such as corrosion were considered negligible and would not affect the roof rupture characteristics for the period of time the covering was absent. This is because the release fasteners are beneath the "Q" decking, and therefore, were provided some protection from the environment. The roof covering installation was completed approximately one and a half months after installation of the roof. Degradation due to corrosion had negligible affect on the release fasteners and no affect on the leakage, as the decking joints were caulked.

The lack of environmental covering on the roof did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) secondary containment integrity had been verified acceptable by testing per the Technical Specification criteria, (2) the margin of safety was maintained without the roof covering, and (3) the boundary conditions for the FSAR evaluations were not changed.

NCR 287-326 (Potential Flooding of Diesel Generators from Automatic Fire Protection Sprinklers)

- Problem Description

  Actuation of all fire protection sprinklers in the High Pressure Core Spray (HPCS) diesel generator room from a fire could potentially flood into the fuel transfer areas for the HPCS and the two standby diesel generator engines, resulting in loss of the fuel transfer pumps. Assuming loss of the HPCS diesel engine from fire, the two standby diesel generators would be lost following consumption of the fuel in the day tanks. This event could potentially result in a common mode failure of all three diesel generators.

  A second postulated common mode failure of the standby diesel generators consists of a fire in the DG-2 room. This would be followed by actuation of the sprinklers resulting in possibly oil and water flooding into the common corridor between the diesel generator and reactor building where Division 1 and 2 conduits are installed. A fire in the corridor could damage the Division 1 cable and result in loss of the safe shutdown capability.

- Corrective Action

  The immediate disposition ("Repair") included the following: (1) install additional curbing at the north exit to all three diesel generator rooms sufficient to preclude flooding in the corridor area; (2) cut slots in each south door of the diesel generator rooms to drain accumulated water to the outside area to prevent flooding of the fuel transfer pumps, and (3) diversion dikes will be placed outside the doors to direct the flow away from the building and air intake areas.

This change is designed to eliminate a common mode failure of the standby emergency diesel generators by eliminating the potential for transfer pump flooding through the addition of the curbing and slots in the south doors to the diesel generator rooms. The second postulated event goes beyond the design bases for the plant, and therefore, is not considered a credible event. However, it should be noted that the curbing would also prevent flooding in the corridor, further reducing the already very unlikely second postulated event. This increases the reliability of the standby diesel generators.

A metal bar was installed across the vertical opening in the south doors to prevent unauthorized persons from entering the diesel generator rooms. Flappers were installed across the openings with magnetic latches to prevent entry of rodents and varmints.

Installation of the curbing and slots in the doors did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) a common mode failure of the standby diesel generators was eliminated increasing their reliability, (2) the margins of safety in the Technical Specifications were unchanged, and (3) the boundary conditions in the FSAR evaluations were not changed.

NCR 288-494 (Lack of Qualification Testing on HPCS Relay Seal-In Unit)

- Problem Description

  A seal-in unit was installed for a High Pressure Core Spray (HPCS) relay (HPCS-RLY-5051/A) without seismic qualification testing. Qualification testing had not been completed on a random sampling of the same lot of seal-in units prior to installation of this seal-in unit. The seal-in unit ensures a constant HPCS relay trip signal until the manual reset is actuated.

- Corrective Action

  The NCR immediate disposition of "Use-As-Is" for this seal-in unit was justified for the following reasons: (1) this seal-in unit design was identical to the one it replaced, and therefore, was expected to pass the seismic qualification testing, and (2) the probability of failure from a postulated seismic event is low because the unit is designed for a high vibration environment.

  Use of the seal-in unit without seismic qualification testing did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the margin of safety provided in the Technical Specifications remained unchanged, and (2) the boundary conditions for the FSAR evaluations were not changed.

NCR 288-538 (Inoperable Fuel Oil Day Tank Transfer Switch)

- Problem Description

  The transfer switch on the top of the fuel oil day tank for the standby diesel generators was bent against the switch cover, making the switch inoperable. Although the inoperable level switch would prevent automatic transfer of fuel oil from the storage tank to the day tank, the low and high level alarms on the day tank would alert operators to provide remote manual control of the transfer pump. It appeared that bending of the switch was caused by using it as a hand-hold to gain access to the top of the tank.

- Corrective Action

  The immediate disposition ("Repair") was to elongate the hole in the top of the switch cover to allow the cover to be offset, and thus, prevent binding of the switch against the cover. The switch is scheduled to be replaced. Also, permanent and easy access to the top of the tank will be provided to eliminate the need to use the switch as a hand-hold.

  This repair did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) operability of the switch was restored, (2) level alarms and remote manual control of the transfer pump provides added assurance the fuel oil day tank would have fuel without overflow under accident conditions, and (3) the boundary conditions in the FSAR evaluation were not changed.

NCR 288-541 (Seat Replacement of Two Containment Exhaust Purge Valves with Unqualified Material)

- Problem Description

  The valve disk seats for two Containment Exhaust Purge (CEP) valves (CEP-V-1A & CEP-V-2A) were replaced November 21, 1985 with a nitrile rubber, tradename Buna-N, instead of the Viton rubber seal material authorized by the CVI data. Although these two valves are physically located in the reactor building (outside of containment), they can potentially experience a containment environment/accident profile internally since they are both located on a 30-inch purge exhaust line that is near the top of the containment.

- Corrective Action

  An analysis was performed to estimate the worst case accident conditions at the valves. The temperature and radiation exposure results of the analysis were compared to the material test data. The test data demonstrated the Buna-N seat material has a service life of six years including the ability to survive the worst case LOCA.

These valves are subjected to periodic technical surveillance requirements. The test surveillance history for the three years of service demonstrated they are functioning as required with the nitrile material. The seats in these valves, as well as all other like valves, are expected to be replaced with a nitrile material within 1989. Therefore, based upon the analysis and test data, the nitrile material is qualifiable for use in the subject valves.

The use of Buna-N rubber instead of Viton did not result in change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the nitrile material is qualifiable for use as seat material in the subject valves, (2) the valve function, performance requirements, and design was not changed, (3) the margin of safety provided in the Technical Specifications remain unchanged, and (4) the boundary conditions for the FSAR evaluations were not changed.

NCR 288-403 (One Train of HVAC Remote Air Intake Valves Disabled and Blocked Open)

o Problem Description

A single failure would cause the Control Room Heating and Ventilation (HVAC) System to operate in the recirculation mode during an emergency condition. An Engineering Evaluation Report, (Design and Operating Deficiencies in Control Room Emergency Ventilation Systems, #E802) was written by the NRC Office for Analysis and Evaluation of Operational Data (AEOD) identifying this generic problem. During a Loss of Coolant Accident (LOCA), the normal fresh air intake for the Control Room HVAC would be isolated and two remote air intake lines would be opened. Each remote air intake line has two isolation valves. One valve on each line is powered by Division 1 and 2. A single failure (hot short was assumed as the failure since the valves are fail open) in a power division could cause a valve in each remote air intake line to isolate. This would result in a neutral pressure condition which would increase the inleakage to the control room, which is an unanalyzed condition.

o Corrective Action

The immediate corrective action ("Repair") was to disable and block open the west remote air intake valves to the Control Room HVAC, i.e., WOA-V-51A and WOA-V-52A. The motor operators on the four remote air intake valves were removed to eliminate electrically induced single failure potentials. One of two purge valves on each intake line were deenergized open to provide more reliable radiation monitoring in the event of the above described failure.

This eliminates the single failure vulnerability of the control room remote air intake system. Both intake paths have Division 1 and 2 operated valves. Without the corrective action, a hot short failure in either division would cause a loss of both remote air intake paths post-accident. Therefore, the reliability of the Control Room HVAC is significantly increased following a LOCA event.

Removal of the motor operators to the four remote air intake valves, blocking open the valves in one remote air intake line, and deenergizing the two intake line purge valves did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the reliability of the remote air intake system following a LOCA event was increased; (2) the operation of the radiation monitors or their ability to detect a plume over a remote air intake was not impacted, (3) the margins of safety in the Technical Specifications were unchanged, and (4) the boundary conditions used in the FSAR evaluations were not changed.

NCR 288-471 (ECCS Relief Valve Setpoints Exceed Piping Design Pressure)

- Problem Description

  A Low Pressure Core Spray (LPCS) relief valve (LPCS-RV-31) setpoint was determined to be set higher than the pump suction piping design pressure. Determination of the wrong setpoint neglected forty feet of static head on the discharge side of the valve. The calculation error was discovered during a SSFI review of the design data base.

- Corrective Action

  The immediate corrective action ("Use-As-Is") was to rely on operational evaluation and actions necessary to prevent an unnecessary pressurization event, which includes opening the LPCS pump suction valve LPCS-V-1 when deemed necessary. The LPCS-RV-31 relief function is required only when the LPCS-V-1 valve is closed. The mechanism for pressurization comes from high to low pressure interface valve leakage.

  There is no Design Basis Accident (DBA) in the FSAR requiring LPCS-V-1 closure. The valve does provide isolation capability, accommodating a single active component failure and is primarily for long term leakage control. However, the bounding Emergency Core Cooling System (ECCS) leakage event determined the largest pump room is filled and concludes that adequate NPSH is available for the remaining ECCS pumps. The ECCS pump room flooding instrumentation will alert the control room of a potential flooding condition, however, the evaluation does not take credit for isolation, i.e., closure of LPCS-V-1. The most probable pressure boundary failure identified for the ECC Systems has been identified as the pump shaft mechanical seal. Valve LPCS-V-1 will be able to be closed given the failure, with less concern for piping protection due to the relief provided by the leak.

The potential for an overpressure condition of approximately 14 psig does not appear from preliminary evaluations to represent a pipe failure condition. In any event, the ECCS pump room flooding analysis bounds a suction piping failure, and therefore, the accident is evaluated in the FSAR.

As a result of this corrective action, the WNP-2 Technical Specifications were not changed and an unreviewed safety question was not involved because: (1) the margin of safety in alerting the control room of either a potential for or an existing intersystem LOCA was not changed, (2) failure of the LPCS pump suction line is bounded by FSAR evaluations, and (3) the boundary conditions for the FSAR evaluations were not changed.

NCR 288-138 (Disabling Corridor Fan in Diesel Generator Building)

- Problem Description

  Under certain emergency conditions, operation of a corridor fan in the Diesel Generator Building, DEA-FN-51, could result in an unmonitored radiological effluent release. During some postulated post-accident conditions, this fan could pull air from the Turbine Building and exhaust it directly to atmosphere. The most severe radiological accident in the Turbine Building is a main steam line break with a source term concentration of 3.312E-4 microCi/cc. The above concentration is within the range specified in Regulatory Guide 1.97 for which effluent monitoring is required.

- Corrective Action

  The immediate corrective action ("Use-As-Is") included: (1) an engineering assessment which determined that DEA-FN-51 was not required for cooling during normal or emergency conditions, (2) disabling the fan by pulling its power fuses, and (3) closing the back draft damper. Further corrective actions will include: (1) removal of the fan DEA-FN-51 and its accessories, (2) sealing the opening created by the removal of the fan, and (3) a design review to ensure that no other potential violations of Regulatory Guide 1.97 due to unmonitored leakage paths exist.

  Disabling the corridor fan and closing the back draft damper did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) a potentially unmonitored, unfiltered radiological release path was eliminated, (2) the margins of safety in the Technical Specifications were unchanged, and (3) the boundary conditions used in the FSAR evaluations were not changed.

NCR 288-389 (Nozzle Ring Setscrew in MSRV Exceed Design Tolerances)

- Problem Description

  The nozzle ring setscrews manufactured for the Main Steam Safety Relief Valves (MSRV) did not meet dimensional tolerances allowed by design. A new setscrew was designed to reduce the failure rate by reducing the thermally induced loads, minimizing stress rises, and increasing the strength of the setscrew.

- Corrective Action

  The immediate corrective action ("Use-As-Is") was to use the as-manufactured setscrews because the as-built configuration is acceptable. The tolerances specified originally were based upon perceived easily achievable tolerances which were much more precise than necessary. The as-built dimensions exceeded the specified allowable tolerances but were within acceptable tolerances for the intended application.

  Using the setscrews with the as-built dimensions which expanded the specified tolerances did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the as-built setscrews do not affect the relief valve setpoints, (2) the margins of safety in the Technical Specifications were unchanged, and (3) the boundary conditions used in the FSAR evaluations were not changed.

NCR 288-395 (Use of Original Setscrew Design for Main Steam Relief Valve)

- Problem Description

  The new nozzle ring setscrews for the Main Steam Relief Valve (MSRV) with the tapered shank could not be installed in one of the MSRVs (MS-RV-2B) due to proximity of other plant equipment. The improvements to the new setscrew included reducing the length to minimize thermally induced loading and tapering the shank to eliminate stress rises in the area where previous failures had occurred.

- Corrective Action

  To allow installation into MS-RV-2B, the tapered shank on the new setscrews was machined down to the original straight shank design. A curved radius was machined at the interface of the smaller straight shank diameter and the larger diameter portion of the setscrew to minimize stress risers. Consequently, the improvements made in the new setscrew were retained in the modified setscrew.

2.6.4 OTHER (Continued)

Using the modified setscrews did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because: (1) the modified setscrews retained the new setscrew design improvements, (2) the margins of safety in the Technical Specifications were unchanged, and (3) the boundary conditions used in the FSAR evaluations were not changed.

2.7 PLANT TESTS AND EXPERIMENTS

Federal Regulations (10CFR50.59) and the Facility Operating License (NPF-21) allow 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.

Prior to performing any test or experiment, a safety evaluation was performed in accordance with 10CFR50.59. All such evaluations were reviewed and approved by the Plant Operations Committee prior to the performance of the tests. It was concluded from the reviews that the tests performed in 1988 did not (1) place the unit in an unanalyzed configuration or condition not bounded by design basis, or (2) perform an operation not described in the FSAR which could have an adverse affect on safety-related equipment or systems. The following are summaries of tests performed in a mode of operation not described in the FSAR. It should be noted, however, that the abnormal mode of operation did not place the unit in an unanalyzed condition.

PPM 8.3.94 & PPM 8.3.10.9 ATWS-ARI Logic Acceptance Test

This procedure was prepared to verify that the Anticipated Transient Without Scram - Alternate Rod Insertion (ATWS-ARI) system installed at WNP-2 meets all of the functional design requirements of the ATWS Rule described in 10CFR50.62 as applicable to boiling water reactors. This logic test verified appropriate operation of the special eight ARI scram air header bleed valves. Loss of air pressure in the scram air header results in insertion of all control rods from the control rod drive accumulators and reactor pressure. The test verified the special valves opened on low reactor level signal or high reactor pressure signal, independent of the Reactor Protection System (RPS). Also, the acceptance criteria that the rods initiate motion within 15 seconds and all rods are fully inserted within 25 seconds from the time of the event was met.

PPM 8.3.90 and 8.3.96 Standby Diesel Engine Governor Control System Preoperational Test

To provide increased reliability of the standby diesel generators, new governor control systems were installed to allow slow start for testing and runback to idle under no load accident conditions. A preoperational test was performed to verify operational capability to runback engine to idle speed under no load conditions preceded by a demand signal and automatic return to full speed on loss of offsite power indication. The test also demonstrated the slow start capability, i.e., start and run engine at idle speed for a predetermined warm up period followed by ramp up to full test speed.

2.8 PLANT PROCEDURE CHANGES

Procedures described in the WNP-2 Final Safety Analysis Report (FSAR) are developed and used by the Plant Operating Staff and various offsite support organizations. In 1988, the Plant Staff made changes to procedures in accordance with 10CFR50.59 and concluded that none of the changes involved unreviewed safety questions.

Changes to procedures were generally either administrative or technical in nature. Administrative revisions consisted of title, organizational and editorial changes; while technical changes were the result of system or component modifications, or improvements in the procedural process. In all instances, a safety evaluation was conducted for each change in accordance with 10CFR50.59. All such evaluations were reviewed and approved by the Plant Operating Committee and are available for audit. It was concluded from the reviews that the probability of occurrence or consequences of an accident or equipment malfunction was not increased, there was no reduction in any plant safety margins, and the possibility of an accident or malfunction not previously evaluated was not increased.

The following is a discussion of significant plant procedure changes and development during 1988:

1. PPM 1.1.7, "Restart Evaluation Process"

   This procedure was developed during the year and provides a process for evaluating the readiness of the plant for startup and return to power following a refueling outage of predefined scope. The process evaluates deviations from the planned outage scope and new work created during the outage. This process, or portions of this process, can also be used at the discretion of the Plant Manager following any major outage. The restart process following a reactor scram is covered separately by PPM 1.3.5, "Reactor Trip and Recovery."

2. PPM 1.2.6, "Biennial Review of Plant Procedures"

   The purpose of this procedure is to describe the review mechanism and provide a reviewer's guide for the biennial review of plant procedures.

   The primary improvement to the procedure was the addition of a procedure reviewer's guide. The guide provides criteria to be used as an aide in performing a quality review of plant procedures. A separate reviewers guide is included for each volume of the Plant Procedures Manual.

3. PPM 1.3.12, "Plant Problems - Problem Evaluation Request"

The purpose of this procedure is to provide instructions for the identification and documentation of plant problems.

This procedure was completely re-written and, used in conjunction with PPM 1.3.15, "Plant Problems - Plant Problem Reports," provides the basis for documenting and resolving both hardware and software problems at WNP-2.

The major change to the procedure is the development of the following four new processes for problem identification and documentation:

- Problem Evaluation Request (PER)

  A document, the purpose of which is to establish a controlled method to formally communicate the existence of a plant problem to plant management for action. This form can be initiated by anyone knowledgeable of an existing or potential plant problem which requires resolution.

- Nonconformance Report (NCR)

  A document, the purpose of which is to disposition all reportable, potentially reportable or safety significant plant problems. This form is initiated by the plant supervisory staff so designated by the Plant Manager.

- Material Deficiency Report (MDR)

  A document, the purpose of which is to disposition all non-reportable, non-safety significant plant problems which directly relate to material, equipment or components (both installed and non-installed). This form is initiated by those members of the Plant supervisory staff or other personnel so designated by the Plant Manager.

- Plant Deficiency Report (PDR)

  A document, the purpose of which is to disposition all non-reportable, non-safety significant plant problems which are not dispositioned by an NCR or MDR. This form is initiated by those members of the Plant supervisory staff so designated by the Plant Manager.

4. PPM 1.3.15, "Plant Problems - Plant Problem Reports"

As previously discussed, this procedure and PPM 1.3.12 form a new framework for documenting and resolving both hardware and software problems at WNP-2. The procedure was developed during the year and provides instructions for the disposition and documentation of plant problems. Feedback to plant personnel of Lessons Learned from inhouse experience is also provided as part of this procedure.

5. PPM 1.3.19, "Housekeeping"

The purpose of this procedure is to provide guidelines and responsibilities to be used to control the cleanliness of WNP-2 facilities.

The primary improvement in this procedure was to expand the responsibilities of the Floor/Area Coordinators in the areas of material deficiencies, industrial safety hazards, cleanliness and housekeeping deficiencies, and radiological protection deficiencies.

6. PPM 1.3.48, "Root Cause Analysis"

The purpose of this procedure is to establish the process and provide instructions for conducting formal analysis of plant events/problems.

The root cause analysis procedure serves to:

- o Improve plant availability by preventing repetitive equipment breakdown through identification of root cause(s) for such failures, and implementation of corrective action which minimizes the probability of recurrence.
- o Provide criteria for evaluation of plant problems such that the level of the problem analysis can be appropriately scaled to the level of the event.
- o Establish a set of organized questions which supplement sound technical judgment in the analysis of plant problems. This concept allows personnel at all levels of the plant organization to participate in implementation of the process.

This is a new procedure developed during the year to formally document the root cause analysis program and it is used in conjunction with PPMs 1.3.12 and 1.3.15 (discussed previously).

7. PPM 1.3.49, "Work Control Center Group"

This procedure was developed during the year and describes the duties of the Work Control Center Group (WCCG) and its interfaces with other plant organizations. The duties include review of all regularly scheduled Maintenance Work Requests (MWRs) to evaluate the need for a Clearance Order (CO). The WCCG also reviews all Preventive Maintenance (PMs) for the need of a Clearance Order. These reviews are performed prior to such work being sent to the Control Room and help in determining the safe configuration of the plant to support necessary maintenance.

The WCCG consists of, but is not limited to, operations representatives (Senior Reactor Operator, Reactor Operator and equipment operator) and maintenance representatives (Electrical and Mechanical).

8. PPM 1.9.1, "Plant Safety Program"

The purpose of this procedure is to outline the WNP-2 Industrial Safety and Fire Protection Program and define the associated responsibilities.

The primary changes to this procedure were adding specific responsibilities of the Plant Safety Marshall and the responsibility for management to perform quarterly reviews of their supervisor's work areas.

The Plant Safety Marshall has the authority to terminate any work which does not meet Supply System Safety requirements.

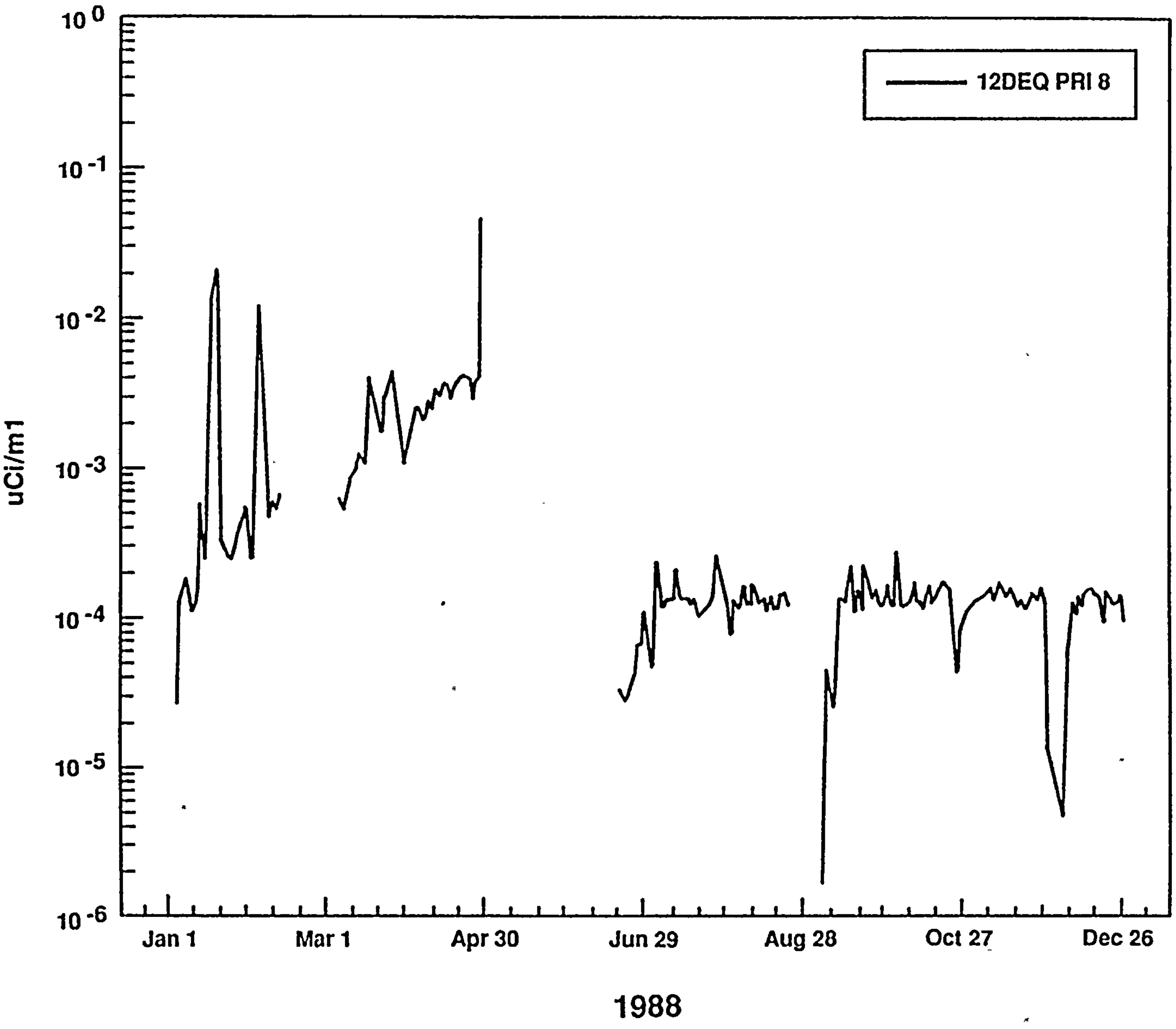
2.9 REACTOR COOLANT SPECIFIC ACTIVITY LEVELS

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

The specific activity of the primary coolant was significantly less than 0.2 microcuries per gram dose equivalent I-131 as set forth in WNP-2 Technical Specification LCO 3/4.4.5 and paragraph 6.9.1.5.c., (see 1988 cumulative iodine graph, attached). The specific activity of the primary coolant was routinely sampled and analyzed as required by WNP-2 Technical Specifications, and was in all cases, less than or equal to $100/\bar{E}$ microcuries per gram.

A graph showing cumulative iodine dose equivalent for the calendar year 1988 is provided for reference and information only. Refer to Section 2.5 of this report for the discussion of the fuel pin failure that caused the iodine trend prior to the 1988 refueling outage.

WNP-2 Reactor Dose Equivalent Iodine



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2.10 REPORT OF DIESEL GENERATOR FAILURES

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

2.10 REPORT OF DIESEL GENERATOR FAILURES

Diesel Generator Failure Number One

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

   Division One Emergency Diesel Generator
   May 22, 1988

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

   Not applicable. This was a nonvalid failure. The unit was inoperable for maintenance and design modification.

3. Cause of failure:

   The direct cause of the failure was an open circuit in the relay K-14A operating coil for the diesel speed control circuit associated with the newly installed idle speed modification. The failure resulted in voltage regulator shutdown which caused generator output to fail. The cause of the relay failure was inadequate design. The relay was an Alternating Current (AC) relay which was installed in a Direct Current (DC) circuit. This resulted in overheating and premature failure of the coil.

4. Corrective measures taken:

   The AC rated relay was replaced with the correct DC rated relay. Both Division One and Division Two idle speed modification designs were reviewed for similar errors.

5. Length of time diesel generator unit was unavailable.

   Not applicable for this nonvalid failure.

6. Current surveillance test interval (after the failure):

   Thirty-one days

7. Verification of test interval:

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

2.10 REPORT OF DIESEL GENERATOR FAILURES (Continued)

Diesel Generator Failure Number Two

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

   Division Two Emergency Diesel Generator
   June 6, 1988

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

   This was the first failure in the last 100 valid tests.

3. Cause of failure:

   The direct cause of the failure was a "relay race" created by inadequate circuit design for the generator exciter control circuit. This circuitry had been recently changed by the idle speed modification installed during the 1988 plant refueling and maintenance outage. The "relay race", rapid operation of the LR relay operating coil, caused overheating and open circuit. This resulted in shutdown of the generator.

4. Corrective measures taken:

   The newly installed design modification was reviewed and changed to eliminate the "relay race" in both Division One and Division Two Emergency Diesel Generators.

5. Length of time diesel generator unit was unavailable:

   Thirty-eight and one half hours.

6. Current surveillance test interval (after this failure):

   Thirty-one days

7. Verification of test interval:

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

Diesel Generator Failure Number Three

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

   High Pressure Core Spray (Division Three) Emergency Diesel Generator
   September 3, 1988

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

   Not applicable. This was a nonvalid failure because it resulted from operator error.

3. Cause of failure:

   The cause of the test failure was the operator failing to assume electrical load rapidly enough to prevent actuation of the reverse power protective relay during initial loading of the unit.

4. Corrective measures taken:

   The operating procedures were modified to add guidance to preclude inadvertent reverse power trips during initial loading of the diesel generators.

5. Length of time the diesel generator unit was unavailable:

   Not applicable for this nonvalid failure.

6. Current surveillance test interval (after this failure):

   Thirty-one days

7. Verification of test interval:

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