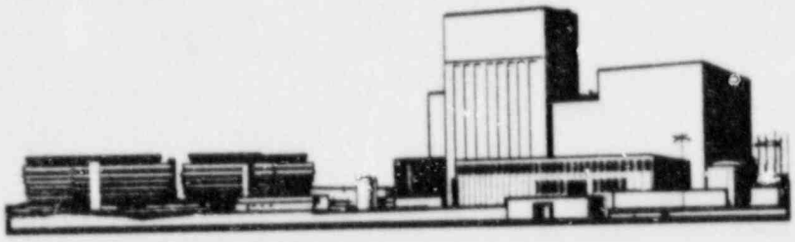
WNP-2 ANNUAL OPERATING REPORT 1987



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM

ANNUAL OPERATING REPORT

OF

WNP-2

FOR 1987

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

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1.0 INTRODUCTION

The 1987 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.

During the first part of 1987 (January-April), the plant continued single-loop operation due to the higher-than-acceptable vibrations experienced on Reactor Recirculation Pump "A". As a result of single-loop operations, power output was limited to 72 percent for that time frame.

On April 10, 1987, the plant was shut down for the annual maintenance and refueling outage. In late June-early July, after the outage, the plant experienced five unplanned automatic shutdowns (scrams) within a two-week period. As a result of those scrams, Supply System management directed that the plant remain shut down pending a thorough review of plant operations. It was concluded from the review that the five scrams were caused by a series of unrelated mechanical and electrical failures that, for the most part, could not have been anticipated.

On July 26, 1987, the plant was restarted and operated for a record 133 consecutive days until December 6, when it was shut down for a scheduled three-day maintenance outage to repair an inoperable flow control valve in the Condensate Filter/Demineralizer System. The inoperable valve had made it necessary to reduce power to 85 percent every four to five days to backwash and precoat the remaining filter demineralizers. Since starting up on July 26, until the planned shutdown on December 6, the plant operated at an average of 91 percent of its capacity. On December 10, the plant was restarted and ran at or near 100 percent capacity for the remainder of the year.

In November, indications of a probable pinhole leak in a single fuel pin were noted. The indications were noted during cycle 3 operation, approximately three days following completion of a control rod sequence exchange, when Off-gas System post-treatment radiation monitor levels began rising. The increase was confirmed by chemical analysis. During a subsequent sequence exchange, selected rod movements were performed in conjunction with on-line pretreatment sample gamma spectroscopy analysis to attempt to isolate the location of the leak. Although initial results appear promising, further testing is planned in conjunction with the next sequence exchange (scheduled for mid-April, 1988) to further isolate/confirm the location of the leak. The current plan is to perform fuel sipping during the upcoming refueling outage (R-3) to identify the leaker and remove it from the core. It should be noted that, to date, no further increases which would indicate more leaking fuel pins have been noted.

During 1987, 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 second refueling outage was successfully completed. Significant activities included:

- o Modification and reinstallation of the two reactor recirculation pumps. Higher-than-acceptable vibrations caused by an internal design defect required that the plant be operated at reduced power since late 1985. Stronger wear rings and bearing assemblies were installed in each pump to solve this problem.

- o Removal of spent fuel assemblies and refueling the reactor. The refueling activity included replacing 148 fuel assemblies, using a fuel shuffle scheme.

- o Removal, inspection, cleaning and reinstallation of two of the three low pressure main turbine rotors.

  - Tube bundles were replaced and improvements made internally to both two-stage Moisture Separator Reheaters.

  - A new moisture preseparator system was installed for excess moisture removal from steam leaving the high pressure turbine.

- o Installation of a new plant process computer. The new computer will enable Operations personnel to more rapidly identify trends that could lead to problems in plant systems. This state-of-the-art system processes information faster than the computer it replaced and has added features such as color graphics display and the ability to provide operators with an historic analysis of an event.

(b) WNP-2 continued to have an excellent record for limiting worker radiation exposure. In 1987, total radiation exposure at the plant was 406 man-rem. Of that total, 98.7 man-rem was attributed to activities associated with repair of the two reactor recirculation pumps. To be rated in the top 25 percent of BWRs for 1987, the Institute of Nuclear Power Operations (INPO) has established the limit of 543 man-rem. In addition, INPO has set 460 man-rem as the industry goal for 1990 for BWRs.

(c) WNP-2 was recognized by the National Academy for Nuclear Training with the accreditation of seven nuclear training programs and the acceptance of the Supply System as a member of the academy. The Supply System is one among 18 of 60 utilities with fully accredited programs.

(d) WNP-2 produced a record 3,315,340 megawatt-hours (gross) of electricity during a record 133 days of continuous operation (July 26 - December 6), at a capacity factor of 91.3 percent. The previous record of a continuous operating period was 100 days (August 4 - November 13, 1985), when 1,918,970 megawatt-hours (gross) was produced.

In October, the plant's best monthly capacity factor of 94.6 percent was achieved, with a record generation of 799,830 megawatt-hours (gross) of electricity.

(e) WNP-2 continued to have a positive trend in the reduction of Licensee Event Reports (LERs). A total of 33 LERs were written during 1987 as compared to 44 LERs in 1986. In addition, the NRC rated the overall average WNP-2 LER score at 8.8, as compared with the industry average of 8.4.

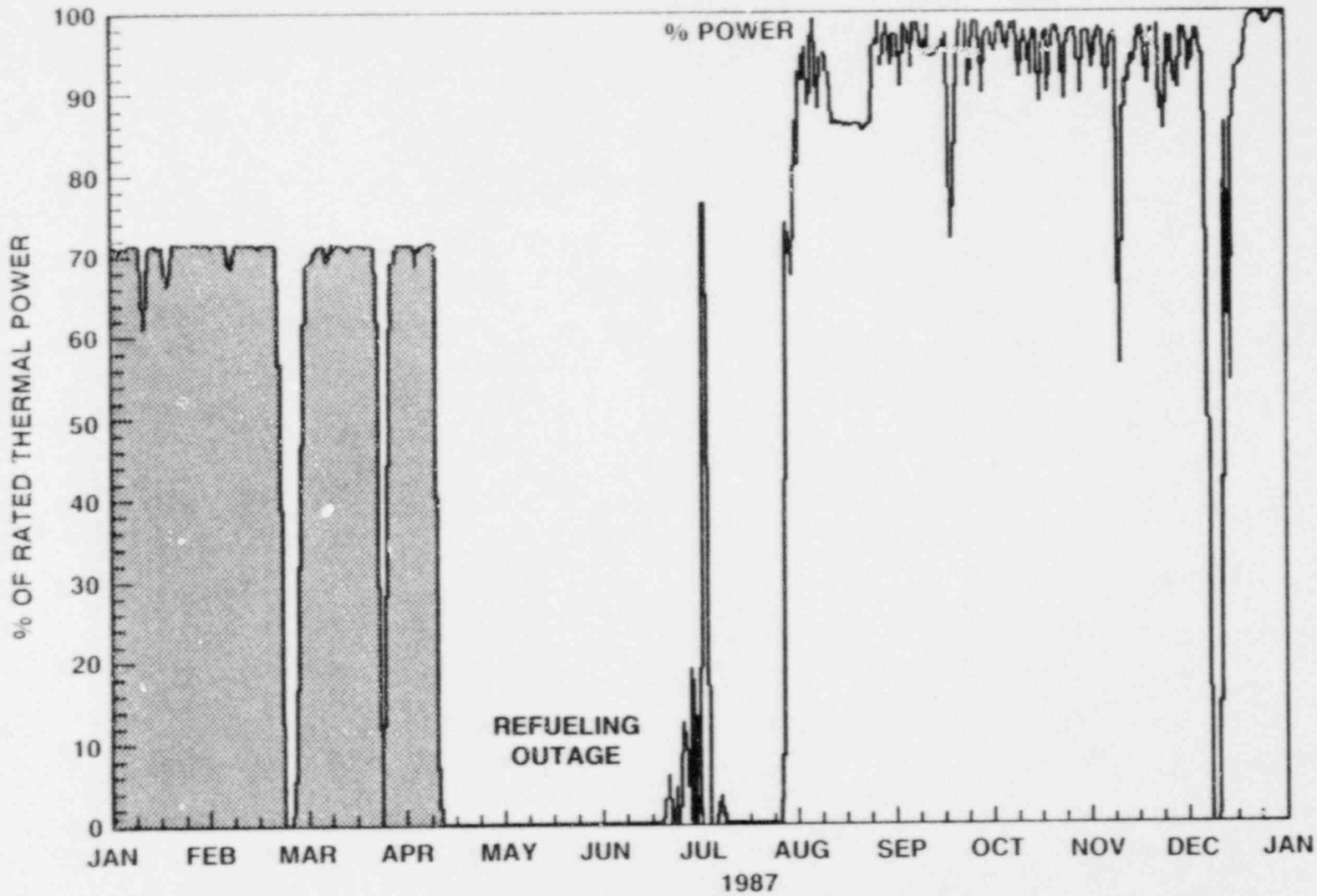
Also during the year, the NRC performed an in-depth Safety System Functional Inspection (SSFI) at WNP-2. The purpose of the inspection was to access the operational readiness of the AC and DC Electrical Distribution Systems, the Standby Service Water System, and the Automatic Depressurization System to function under all operational and analyzed accident conditions. At the conclusion of the inspection, it was noted that no significant deficiencies were identified which would prevent the reviewed systems from performing their intended functions. However, it was concluded that improvements were needed in several areas. In the inspection report, the inspectors identified ten Notices of Violation (five at Level IV and five at Level V). As a result, the Supply System has developed an action plan for strengthening the design modification process. The action plan addresses the broader programmatic issues which either were identified by the NRC SSFI team, or which resulted from Supply System consideration of underlying causes which may have contributed to the deficiencies noted by the NRC. The action plan includes both near and long-term initiatives to address the programmatic improvement needs as a result of the SSFI. In addition, we will continue to monitor our performance in this area and make improvements where necessary.

The 1987 actual and adjusted capacity factors, based upon net electrical energy output, are listed in the following table. For the period of time when the plant was operating with one recirculation pump, the adjusted capacity factor was based on a maximum power output of 71.7 percent rather than 100 percent.

Month		Capacity Factor	Adjusted Capacity Factor
January	***	69.6	97.1
February	***	55.0	76.6
March	***	63.4	88.3
April	*	20.8	29.0
May		0	0
June	**	1.8	1.8
July		15.3	15.3
August		89.2	89.2
September		93.1	93.1
October		94.6	94.6
November		91.9	91.9
December		79.8	79.8
Overall		56.2	63.0

* Started Maintenance/Refueling Outage
** Ended Maintenance/Refueling Outage
*** Single Loop Operation

WNP-2
1987 POWER HISTORY



☐ MAXIMUM POWER OUTPUT LIMITED TO APPROXIMATELY 72% BASED ON SINGLE LOOP OPERATION

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.

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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.

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

NUCLEAR PLANT NO 2

REPORT FOR CALENDAR YEAR 1987

		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	26.356	0.083	47.317	9.726	0.009	22.591
	OPERATING PERSONNEL	27.746	0.000	0.000	15.058	0.000	0.000
	HEALTH PHYSICS PERSONNEL	22.466	0.875	8.403	14.777	0.153	4.238
	SUPERVISORY PERSONNEL	10.077	2.727	1.813	4.731	1.183	0.223
	ENGINEERING PERSONNEL	6.395	17.522	16.749	3.805	6.398	6.746
ROUTINE MAINTENANCE	MAINTENANCE PERSONNEL	73.814	0.299	62.680	46.007	0.036	24.231
	OPERATING PERSONNEL	6.777	0.402	0.000	3.203	0.089	0.000
	HEALTH PHYSICS PERSONNEL	10.417	0.105	13.539	13.506	0.018	19.530
	SUPERVISORY PERSONNEL	0.526	0.141	0.034	0.257	0.027	0.004
	ENGINEERING PERSONNEL	6.192	4.770	4.041	3.138	1.296	1.255
INSERVICE INSPECTION	MAINTENANCE PERSONNEL	6.809	0.000	4.633	2.245	0.000	2.969
	OPERATING PERSONNEL	0.299	0.040	0.000	0.152	0.009	0.000
	HEALTH PHYSICS PERSONNEL	0.391	0.000	0.515	0.481	0.000	0.210
	SUPERVISORY PERSONNEL	0.752	0.271	0.000	0.112	0.107	0.000
	ENGINEERING PERSONNEL	1.263	1.366	1.978	0.264	0.381	0.631
SPECIAL MAINTENANCE	MAINTENANCE PERSONNEL	109.112	0.000	87.492	93.545	0.000	32.050
	OPERATING PERSONNEL	1.920	0.558	0.000	1.129	0.124	0.000
	HEALTH PHYSICS PERSONNEL	5.514	0.000	21.403	7.376	0.000	20.198
	SUPERVISORY PERSONNEL	0.685	1.825	1.000	0.485	0.694	0.891
	ENGINEERING PERSONNEL	8.590	6.528	6.960	3.563	3.008	3.640
WASTE PROCESSING	MAINTENANCE PERSONNEL	4.530	0.000	0.000	3.454	0.000	0.000
	OPERATING PERSONNEL	0.004	0.000	0.000	0.004	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.738	0.000	0.265	0.639	0.000	0.367
	SUPERVISORY PERSONNEL	0.000	0.000	1.000	0.000	0.000	3.768
	ENGINEERING PERSONNEL	0.041	0.460	0.000	0.018	0.081	0.000
REFUELING	MAINTENANCE PERSONNEL	7.995	0.000	0.037	5.314	0.000	0.007
	OPERATING PERSONNEL	1.254	0.000	0.000	0.603	0.000	0.000
	HEALTH PHYSICS PERSONNEL	0.538	0.000	0.810	0.586	0.000	0.519
	SUPERVISORY PERSONNEL	1.108	0.036	0.154	0.613	0.009	0.018
	ENGINEERING PERSONNEL	1.171	0.354	0.110	0.323	0.125	0.049
TOTAL	MAINTENANCE PERSONNEL	228.616	0.382	202.159	160.291	0.045	81.860
	OPERATING PERSONNEL	38.000	1.000	0.000	20.149	0.222	0.000
	HEALTH PHYSICS PERSONNEL	40.064	1.000	44.935	37.368	0.171	36.062
	SUPERVISORY PERSONNEL	13.148	5.000	4.001	6.198	2.020	4.904
	ENGINEERING PERSONNEL	23.852	31.000	29.838	11.111	11.289	12.321
GRAND TOTAL		343.680	38.382	280.933	235.117	13.747	135.147

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information concerning main steam line safety/relief valve challenges for calendar year 1987 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
03/22/87	MS-RV-5B	B	G	E	0%	87-002
03/22/87	MS-RV-5C	B	G	E	0%	87-002
03/22/87	MS-RV-3D	B	G	E	0%	87-002
03/22/87	MS-RV-4D	B	G	E	0%	87-002

The 03/22/87 actuations were in response to a plant trip.

04/10/87	MS-RV-1A	B	D	C	20%	--
04/10/87	MS-RV-2A	B	D	C	20%	--
04/10/87	MS-RV-3A	B	D	C	20%	--
04/10/87	MS-RV-4A	B	D	C	20%	--
04/10/87	MS-RV-1B	B	D	C	20%	--
04/10/87	MS-RV-2B	B	D	C	20%	--
04/10/87	MS-RV-3B	B	D	C	20%	--
04/10/87	MS-RV-4B	B	D	C	20%	--
04/10/87	MS-RV-5B	B	D	C	20%	--
04/10/87	MS-RV-1C	B	D	C	20%	--
04/10/87	MS-RV-2C	B	D	C	20%	--
04/10/87	MS-RV-3C	B	D	C	20%	--
04/10/87	MS-RV-4C	B	D	C	20%	--
04/10/87	MS-RV-5C	B	D	C	20%	--
04/10/87	MS-RV-1D	B	D	C	20%	--
04/10/87	MS-RV-2D	B	D	C	20%	--
04/10/87	MS-RV-3D	B	D	C	20%	--
04/10/87	MS-RV-4D	B	D	C	20%	--

The 04/10/87 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
06/18/87	MS-RV-3B	B	G	C	0%	--
06/18/87	MS-RV-3B	B	G	C	0%	--
06/18/87	MS-RV-4A	B	G	C	0%	--
06/18/87	MS-RV-4A	B	G	C	0%	--
06/18/87	MS-RV-2A	B	G	C	0%	--
06/18/87	MS-RV-2A	B	G	C	0%	--
06/18/87	MS-RV-4B	B	G	C	0%	--
06/18/87	MS-RV-4B	B	G	C	0%	--
06/18/87	MS-RV-2C	B	G	C	0%	--
06/18/87	MS-RV-2C	B	G	C	0%	--
06/18/87	MS-RV-4C	B	G	C	0%	--
06/18/87	MS-RV-4C	B	G	C	0%	--

The 06/18/87 actuations verified S/RV operability from remote and alternate remote shutdown panels. This has since been incorporated into acoustic monitoring surveillances.

06/20/87	MS-RV-4D	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-2A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-2A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-3A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-3A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4A	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-3B	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-3B	C + Hydroset	C	C	3%	--

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
06/20/87	MS-RV-5B	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-5B	C + Hydroset	C	C	3%	--
06/20/87	MS-RV-4D	C + Hydroset	C	C	3%	--

The 06/20/87 actuations were part of the setpoint verification surveillance.

06/24/87	MS-RV-1A	B	C	C	10%	--
06/24/87	MS-RV-2A	B	C	C	10%	--
06/24/87	MS-RV-4A	B	C	C	10%	--
06/24/87	MS-RV-1B	B	C	C	10%	--
06/24/87	MS-RV-3B	B	C	C	10%	--
06/24/87	MS-RV-1D	B	C	C	10%	--
06/24/87	MS-RV-3D	B	C	C	10%	--
06/24/87	MS-RV-4B	B	C	C	10%	--
06/24/87	MS-RV-5B	B	C	C	10%	--
06/24/87	MS-RV-1C	B	C	C	10%	--
06/24/87	MS-RV-4C	B	C	C	10%	--
06/24/87	MS-RV-5D	B	C	C	10%	--

The 06/24/87 actuations were performed to reduce leakage through the valve seats.

07/02/87	MS-RV-1B	A	G	A	0%	87-020
07/02/87	MS-RV-1B	B	G	E	0%	87-020
07/02/87	MS-RV-1C	A	G	A	0%	87-020
07/02/87	MS-RV-2C	B	G	E	0%	87-020
07/02/87	MS-RV-3B	A	G	A	0%	87-020

The 07/02/87 actuations were in response to a reactor trip.

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
11/26/87	MS-RV-2D	A	E	D	97%	--
11/26/87	MS-RV-2D	A	E	D	97%	--

The 11/26/87 actuations were caused by a plant technician incorrectly installing a set of jumpers.

12/11/87	MS-RV-2D	B	E	C	15%	--

The 12/11/87 actuation was initiated to verify a repair on the associated acoustic monitor.

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
11/10/86 thru 4/10/87	F	0	A	5	--	CB	PUMPX	Power output limited to 72% due to inoperability of the "B" Recirculation Pump.
2/20/87	F	122	A	1	--	HA	TURBIN	The plant was shut down because of low turbine bearing oil pressure. An inspection revealed a broken check valve in the bearing oil supply header.
2/25/87	F	5.9	G	1	--	HA	GENERA	The turbine generator was removed from service due to high exciter temperature which initiated the fire protection system and admitted CO_2 inside the exciter housing. An inspection revealed that the cooling water valving was misaligned. The exciter was inspected for damage and the plant was 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
3/22/87	F	57.3	A	2	87-02	CH	INSTRU	The reactor was manually scrammed at 71% power due to a sudden reduction in RPV level caused by loss of both feedwater pumps on low suction pressure. A failed fuse in the feedwater level controller caused the event. Recovery efforts led to the flooding of the Main Steam Lines up to the closed MSIVs, due to the improper lineup of the RFW System startup level control valve. The plant was brought to cold shutdown. An engineering evaluation was performed to include extensive plant piping and support system inspections to verify no adverse effects.
4/10/87 thru 6/25/87	S	1823	C	1	--	RC	FUEL	The plant was shut down as scheduled for the annual refueling outage.
6/25/87	S	10.73	B	1	--	HA	MECFUN	The generator was removed from the grid to perform turbine overspeed testing.

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
6/26/87	F	17.63	A	3	87-18	EB	TRANSF	The reactor automatically scrammed due to a Reactor Protection System (RPS) actuation from a Turbine Control Valve (TCV) fast closure. The TCV fast closure was in response to a unit lockout signal, found later to be the result of a sudden pressure relay actuation on the Normal Auxiliary Power Transformer, TR-N1. Following the 6/26 scram, the plant was restarted with TR-N1 out of service for further evaluation. The cause of the 6/26 scram was thought to be a degraded TR-N1 transformer that had previously sustained potential degrading transients and showed signs of recent degradation from oil samples taken.
6/27/87	F	26.32	A	3	87-18	EB	TRANSF	The same sequence of events occurred on 6/27, causing a similar reactor scram due to TR-N2. The cause of both scrams was determined to be the opening of a (poppet) valve used to test the relay trip function. Following the scram on 6/27, the poppet valves were removed and plugged to prevent recurrence.

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
6/28/87	F	29.13	A	3	87-19	EB	CKTBKR	The reactor scrammed at 54% power due to a spurious loss of Motor Generator (MG) power to RPS Bus A, coincident with an existing RPS 1/2-trip from an inoperable/tripped Governor Valve fast closure pressure switch. The cause of the RPS Channel "A" breaker tripping appeared to be the improper alignment of the undervoltage restraint coil. The circuit breaker and oil pressure switch were replaced and tested prior to plant restart. Subsequent analysis by General Electric suggested another probable cause of coil failure was thermal aging.
7/2/87	F	88.03	A	3	87-20	EB	GENERA	The reactor scrammed from 80% power due to a loss of power to both RPS buses. Initially a Motor Generator (MG) Set failed, causing the loss of RPS "A" power. When plant operators attempted to switch RPS "A" power to its alternate source, per procedure, RPS "B" power was lost. The deenergization of both RPS buses causes, by design, a reactor scram. The faulty MG Set motor was replaced and tested. An inspection of the transfer switch, common to both RPS divisions of power, revealed a broken stop tab which was repaired and tested.

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
7/6/87	F	.27	A	9	--	HA	RELAYX	Two minutes after synchronizing, the generator tripped due to insufficient load pickup by the DEH System. After evaluation, a higher load reference and rate was used and the generator was resynchronized successfully.
7/6/87	F	223.23	A	3	87-22	EB	CKTBKR	The reactor scrammed from 33% power on low reactor pressure vessel (RPV) water level. During the transfer of plant electrical loads from the Startup Power Supply to the Normal Power Supply, an electrical breaker failed, resulting in a loss of the inservice Reactor Feedwater (RFW) pump and subsequent loss of feedwater to the RPV. The cause of the breaker failure was a bent finger in the "C" Phase Disconnecting Contact Finger Cluster, which prevented the breaker from fully inserting. The breaker was repaired and extensive testing was performed prior to plant restart.

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
7/15/87	S	267.47	B	9	--	--	--	Supply System management directed that the plant remain shut down pending a thorough review of the events surrounding the five previous scrams. A subsequent Confirmatory Action Letter concurred with Supply System management's decision and outlined steps to be taken prior to reactor startup. Areas evaluated include, but were not limited to, 1) an evaluation of the effectiveness of the WNP-2 post-trip review and root cause assessment programs, 2) a reevaluation of the root cause analysis of problems encountered during the recent startup program, 3) an assessment of major work items accomplished during the 1987 refueling outage, and 4) discussions with members of plant staff regarding the implementation of the root cause analysis program. At the conclusion of this intensive review, programmatic changes and corrective actions were discussed with the NRC and found to be acceptable, resulting in plant restart.
9/17/87	S	0	H	5	--	RB	CONROD	Reactor power was reduced, as required, to perform a scheduled control rod sequence exchange.

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
11/9/87	S	0	H	5	--	RB	CONROD	Reactor power was reduced, as required, to perform a scheduled control rod sequence exchange.
12/6/87	S	103.8	B	1	--	--	--	The plant was shut down in support of various inspection and repair activities necessary to support sustained rated power operation. Activities included inspection of condensate filter demineralizers (CFD), repair of CFD outlet valves, replacement of resin strainers, repair of reactor building ventilation supply fans and correction of secondary side steam leaks.
12/13/87	S	4.1	B	1	--	--	--	The generator was removed from service to perform tests to evaluate a mechanical binding condition of turbine governor valve #4.

CAUSE CODE	TOTAL FOR 1987	TOTAL GENERATOR OFF-LINE HOURS
A	9	563.9
B	4	386.1
C	1	1823.0
D	0	0.0
F	0	0.0
G	1	5.9
H	2	0.0
TOTAL		2778.9

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
CF	Residual Heat Removal Systems & Controls
CH	Feedwater Systems & Controls
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
HJ	Other Features of Steam & Power Conversion Systems (not included elsewhere)
MS	Main Steam System
RB	Reactivity Control Systems

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
Instrumentation and Controls (INSTRU)	Controllers Sensors/Detectors/Elements Indicators Differentials Integrators (Totalizers) Power Supplies Recorders Switches Transmitters Computation Modules
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

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

COMPONENT CODE (continued)

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
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
CRD-TK-125/2203 CRD-TK-125/2247 CRD-TK-125/4243 CRD-TK-125/1051	Control Rod Drive	Prior inspections have indicated signs of corrosion on the internal walls of these water accumulators.	Removed water accumulators and replaced with new spares. Performed a leak test on all connections.
SW-V-90	Service Water	Valve motor runs, but the valve stem does not move.	Disassembled valve and found a broken shaft assembly. Replaced broken parts, greased, tested and returned to service.
CRD-DRVE-5431 CRD-DRVE-1803	Control Rod Drive	Control Rod Drives have no position indication at notch "46".	Removed and replaced position indicator probes per procedure and verified position indication at all notches.
REA-FT-7 REA-SQRT-1	Reactor Building Exhaust Air	Elevated release stack flow instrument failed to respond during the performance of a surveillance test.	Replaced the associated flow transmitter and square rooter and calibrated the instruments. Reperformed applicable steps of surveillance procedure and returned to service.
DLO-TS-34B1	Diesel Lube Oil	The temperature switch for DG "B" Engine Oil cooler outlet appears to be giving false indication.	During troubleshooting activities, the temperature switch was found to be damaged. The switch was replaced and recalibrated per plant procedures.

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

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
MSLC-PS-20	Main Steam Leakage Control	The operational trending program indicates that MSLC-PS-20 switch #2 is resetting erratically.	Removed instrument from service and replaced the switch with a new spare. Recalibrated the instrument per plant procedure and returned to service.
TEA-SR-38	Turbine Exhaust Air	Particulate and iodine flow control is malfunctioning.	Secured sample rack and replaced flow control valve #2 with a spare. Replaced the diaphragm on flow control valve #1. Performed restoration of sample rack per plant procedure.
DLO-P-6	Diesel Lube Oil	HPCS Diesel Circulating Oil Pump coupling failed.	The pump was taken out of service to replace a broken coupling. Upon completion of the work, the pump was returned to service.
DSA-LOC-4B2/1	Diesel Starting Air	A left-handed air lubricator was installed by a vendor on the right-hand air line (i.e., air flow is backwards). Replace lubricators with the correct type.	Removed reservoir from replacement lubricator and cleaned with solvent. Isolated the air header and removed old lubricator unit. Replaced the installed lubricator with the new lubricator and returned to service.
DO-LS-12A	Diesel Oil	The low-level switch on day tank #3 is shorted to the housing cover.	Removed the electrical cover and switch housing and cleaned them as much as possible. Verified that the switch was still operational. Repaired damaged wiring per plant procedure, recalibrated instrument and returned to service.

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

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
MS-AO-22B MS-AO-22C	Main Steam	General Electric requires a five-year Main Steam Isolation Valve operator overhaul.	Valve operators were removed from valves and moved to a suitable work area. A sample of hydraulic fluid was taken and all elastomers were logged for wear analysis and potential EQ life extension. Leakage tests were performed and found to be acceptable. The operators were reassembled and installed on the valves.
MS-SPV-22A1, 2, 3 MS-SPV-22B1, 2, 3 MS-SPV-22C1, 2, 3 MS-SPV-22D1, 2, 3 MS-SPV-28A1, 2, 3 MS-SPV-28B1, 2, 3 MS-SPV-28C1, 2, 3 MS-SPV-28D1, 2, 3	Main Steam	The equipment qualification program requires periodic maintenance of the main steam isolation valve solenoid valves. Replace coils and elastomers.	Isolated electrical and air systems and disconnected leads from each solenoid. Replaced coils and valve seals with environmentally qualified spares. Reassembled the valves and returned to service.
RPS-PS-5D	Reactor Protection System	The Turbine Generator Valve fast closure pressure switch was leaking DEH oil.	Removed pressure switch from service and replaced with a new spare. Recalibrated the instrument and performed a channel check, returned the instrument to service.
LPRM-DET-08/25	Low Power Range Monitor	Two channels appear to have erroneous output - investigate and repair as necessary.	During troubleshooting activities APRM channel C was found to have a defective power supply in the ion chamber. Replaced the transformer and tested satisfactorily. The other channel was tested and appears to be operating normally.

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

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
LPCS-42-7B6B	Low Pressure Core Spray	The motor starter for the LPCS water leg pump appears to be broken.	Troubleshooting activities found that the motor starter had some auxiliary contacts broken off. The magnetic starter was replaced, all connections were checked for tightness and the operational test was successfully completed.
RCC-V-6	Reactor Closed Cooling Water	The valve closes and cannot be electrically opened.	A loose wire was found on a terminal block in the motor control center. A broken torque switch was also found on the valve. The wire was tightened and the torque switch replaced and the valve tested satisfactorily.
RPS-EPA-3C	Reactor Protection System	The reactor scrammed due to the opening of RPS MG Set breaker.	Removed EPA breaker 3C and inspected the undervoltage restraint coil. Found slight armature bending. Adjusted the coil to relieve the bending and exercised the trip mechanism to verify proper operation. Also inspected and verified operation of breakers 3A, B, D, E and F.
TEA-SR-38	Turbine Exhaust Air	The sample rack for Turbine Building stack flow shows poor linear flow characteristics. Remove, inspect and repair as necessary.	The isokinetic flow valve was cleaned, adjusted. The associated flow switch, transducer and square rooter were replaced during overhaul of the sample rack.

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

EQUIPMENT REQUIRING MAINTENANCE	SYSTEM	PROBLEM	ACTION TAKEN
CMS-TI-43R	Containment Monitoring System	Suppression pool water temperature instrument on the remote shutdown panel failed high.	Repaired a splice for the temperature element in terminal box TBC-501. A wire in the outboard penetration box was also found damaged and was respliced in accordance with plant procedures.
REA-E/S-613C	Reactor Building Exhaust Air	The Reactor Building exhaust plenum radiation monitor has failed downscale.	During troubleshooting, technician found no 24-volt supply to the instrument. One bad fuse was found and replaced. A functional test was performed and the instrument was returned to service.

2.4 OTHER SIGNIFICANT MAINTENANCE ITEMS

CRD Removal/Replacement

Twenty (20) control rod drive mechanisms were removed from the reactor vessel and replaced with rebuilt spares. The drives were selected for maintenance based upon trended performance characteristics such as stall flows, high temperature, difficulty notching out of "00" or other operational problems.

Turbine Maintenance

During the 1987 refueling outage, the main turbine was disassembled for manufacturers' required inspections. Low pressure (LP) turbines #1 and #2 were completely disassembled and inspected for signs of deterioration. A disc crack indication was found on LP #1. Due to this failure, the Supply System shortened the frequency of inspection from 47 months to 31 operating months for this component. No significant problems were found on LP #2. Two throttle valves were partially disassembled/inspected, two reheat stop valves and two intercept valves were also inspected with no significant problems identified. All governor valves and bypass valves were completely disassembled and refurbished. New wear rings were installed on the governor valves and the bypass valves needed various internal components. Eight new Moisture Separator Reheater (MSR) tube bundles were installed due to the presence of rust, tube failure and support plate bowing on the originally installed tube bundles.

Reactor Recirculation Pump Modification

A significant portion of 1986 and 1987 was spent in single loop operation due to reactor recirculation pump problems. During the 1987 refueling outage, both recirculation pumps were overhauled and modified. Modifications to pump 1A included strengthening of the overall bearing design as well as a reinforcement of the wear ring surfaces. Pump 1B had the bearing design modification implemented during 1986; therefore, only reinforcement to the wear ring was necessary during 1987. The repairs implemented during this refueling outage have successfully solved this problem and the unit has been able to set new generation records in double loop operation.

Limitorque Motor Operator Torque Switch Bypass

During investigation into the failure of a Main Steam Leakage Control (MSLC) motor operator, it was noted that a required torque switch jumper was not installed. Documentation was found indicating that jumpers had been correctly installed on the valve during the startup phase of construction. An engineering analysis was performed and indicated the potential for other errors with regard to missing torque switch jumpers. Based upon this analysis, the decision was made to perform field inspection of all MOVs having an "open" safety function. Of the 66 MOVs inspected, 14 were found missing the required jumpers. All required jumpers were installed and top tier drawings were verified to be correct or were modified as necessary.

2.4 OTHER SIGNIFICANT MAINTENANCE ITEMS (Continued)

Motor Control Center Configuration Control

As a result of the Limitorque MOV deficiencies, a fuse and motor overload relay inspection was conducted to verify that the plant configuration complies with design requirements. Of the more than 700 circuits inspected, 83 discrepancies were identified. Sixty-two (62) circuits could not be adequately inspected with the related equipment energized. Twenty-eight (28) of the 83 discrepancies have subsequently been corrected, and 40 of the 62 circuits previously unavailable for inspection have been successfully inspected. The remaining inspections and corrections will be completed either during or before the next refueling outage. Also, a procedure controlling fuse replacement has been drafted for implementation in 1988 to enhance our ability to control plant configuration relative to fuses.

Installation of Viton Seals

During 1987, all ITT General Controls hydromotor polyurethane seals installed on WNP-2 valve actuators were replaced with Viton seals. This modification was made because, with the originally installed seals, the actuators could not be environmentally qualified for the six-months post LOCA as required by the FSAR. Forty-seven (47) class 1 actuators, ten nonsafety-related and all spares located in the warehouse were upgraded. This modification will extend the service life on each of the actuators an average of eight years.

Safety-Related MOV Inspection and Upgrade

Due to a number of equipment qualification group concerns, an inspection of safety-related motor operated valves was performed. Items of interest during the inspection included: properly insulated splices, installation of correct terminal blocks, valve orientation relative to drain plugs, integrity of torque switches and correct use of space heaters. Various deficiencies were corrected and a large number of cables were respliced using Raychem shrinkable materials. The addition of this insulation has resulted in a greatly increased effective service life of these valves.

Three-Year Diesel Engine Maintenance

The Supply System performed the manufacturers' recommended one-year and three-year preventive maintenance to all five of the WNP-2 diesel engines. This maintenance included, but was not limited to, the inspection of the turbocharger and exhaust piping, replacement of air-start motors, rebuilding or replacement of starting air system solenoid valves, inspection of selected main and connecting rod bearings, replacement of all filter elements in the lube oil, fuel oil and intake air systems and engine one-revolution inspection. All equipment was found to be free of excessive wear or degradation.

Implementation of MOVATS Program

During 1987, the Supply System responded to IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," by implementing the MOVATS Program.

Site Engineering personnel determined which valves would be tested in accordance with 10CFR50.55a(g). A program was then developed to ensure that the selected valve operators were tested and subsequently properly maintained. As a result of testing, several deficiencies were noted, including torque switches not bypassed in the opening direction and torque switches set too high. Other items inspected were overall valve integrity and correct limit switch settings. In addition to meeting the IE Bulletin requirements, the Supply System has decided to expand MOVATS testing to other safety-related valve operators.

All deficiencies were corrected and an engineering evaluation was performed to ensure that all valves were not overstressed due to high torque switch settings. All valves have been approved for use until the 1988 refueling outage and further determination for long-term action is anticipated at that time.

2.5 INDICATIONS OF FAILED FUEL

In accordance with the WNP-2 FSAR, Section 4.2.4.3, a visual inspection of discharged fuel assemblies from Cycle 2 was performed. This information is supplied in accordance with requirements as set forth in Regulatory Guide 1.16.

Selection of Assemblies

Eight assemblies were selected for inspection representing greater than 5 percent of the discharged fuel. The selected assemblies are all medium enriched (1.76 w/o U-235). Some characteristics of the selected assemblies are shown below.

CYCLE 2 DISCHARGED FUEL ASSEMBLIES
SELECTED FOR EXAMINATION

Fuel Assembly	Haling Radial Power	Cycle 2 Core Location	POWERPLEX EOC 2 Exposure	Bundle Type
Assembly Group 1				
LJT 157	1.108	37-40	14,266	GE 1.76
LJT 159		23-22	14,265	GE 1.76
LJT 163		37-22	14,266	GE 1.76
LJT 164		23-40	14,266	GE 1.76
Assembly Group 2				
LJT 259	.968	21-10	14,338	GE 1.76
LJT 321		21-52	14,338	GE 1.76
LJT 332		39-52	14.339	GE 1.76
LJT 357		39-10	14.338	GE 1.76

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. Thirty-five mm (35mm) photographs were taken of the points of interest. A total of 52 photographs of the examined fuel were taken. Of these, 45 were successful photographs. 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 by means of the fuel-handling mast on the refuel bridge.

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,
- Relative rod growth,
- Condition of tie rod hex nuts and other structural components,
- Nodular corrosion and crud formation, and
- Rod fretting.

Results of the Examination

The inspected fuel assemblies appeared to have good structural integrity. The upper tie plates were level, rod springs had ample compression space, tie rod nuts were snug, there was no apparent rod bow and all rods were properly seated in the lower tie plate. The spacers were perpendicular to the rods and properly located. Major damage to one set of finger springs was noted.

Small scratches attributed to dechanneling were observed on some peripheral rods and on spacer surfaces. There was no evidence of debris damage. There was also no evidence of fretting behavior. In general, the mechanical integrity of the fuel appears to be good.

The rate of crud and nodular buildup on the fuel appears to have increased from that observed after Cycle 1. Some of the nodules seem to have started to come together to exhibit a spallation type of appearance. On one bundle, the crud layer, which was quite heavy, was cleaned from one fuel rod by mechanical cleaning. Oxide nodules were observed on this fuel rod where the crud had been.

The grid spacers exhibit a very heavy nodular buildup. In addition, several of the spacers exhibit holes in the heat-affected zone of the welds. These holes are not an isolated phenomena. One spacer had four holes on one side and three on another. There is some disagreement at present as to the origin of these holes, with some reviewers believing they are a corrosion-based phenomena and others believing they were caused at manufacture.

Summary

The inspected fuel appears to be free of significant mechanical damage caused by plant operation. Mechanical damage appears to be limited to minor scratches except for damage to one set of finger springs which was felt to have occurred during handling after fuel discharge.

Heavy nodular corrosion exists on virtually all spacers. Significant nodular corrosion was observed on many of the fuel rods and is present under the crud layer. Some fuel regions appear to be exhibiting a spallation-type growth of nodules. The overall growth of nodule deposit appears to be at least keeping pace with the fuel exposure.

Based on the observed data, the risk of fuel failure from nodular corrosion appears to be low for Cycle 3. However, the current inspection techniques do not allow for examination of the Gd_2O_3 fuel rods which are usually the fuel rods at most risk to this type of fuel failure.

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 1987 are provided. Included are summaries of the safety evaluations.

2.6.1 PLANT DESIGN CHANGES

The following plant design changes were completed in 1987 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.

PLANT DESIGN CHANGE 83-0047

Plant Design Change 83-0047 was initiated to accommodate surface drainage during severe conditions, such as rainfall and rapid snowmelts, by installing a system of catch basins. This modification also included the final site paving and grading plan.

This plant design change provided direction for the installation of a system of catch basins to provide adequate drainage during conditions of severe water or snow accumulation. This system replaces the previously used system of ditches and culverts and completes the deferred construction item of final site paving and grading.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the modification actually precludes the potential flooding of safety-related structures as evaluated in FSAR Section 3.4.1.4.1.1.

PLANT DESIGN CHANGE 84-1071

Plant Design Change 84-1071 was initiated to provide a dedicated back-up water supply for the Fire Protection System.

This modification installed a 400,000-gallon bladder tank which serves as the secondary source of water for the Fire Protection System. The system previously used was not a dedicated source and also had a smaller storage capacity.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the margin of safety is increased by the additional water supply for fire protection.

PLANT DESIGN CHANGE 84-1144

Plant Design Change 84-1144 was initiated to design and install a decontamination facility within the Radwaste Building of the plant.

This modification package included design for modifications of support equipment such as the HVAC system, floor drains, chemical waste and electrical systems, as well as the placement of essential decontamination equipment within the decon room.

This modification did not result in a change to WNP-2 Technical Specifications or involve an unreviewed safety question because the equipment being added is not safety-related and because structural, HVAC and electrical loads associated with this modification are within the design rating of equipment and plant structures.

PLANT DESIGN CHANGES 84-1249 AND 84-1322

Plant Design Changes 84-1249 and 84-1322 were initiated to remove the gaseous chlorination portion of the plant make-up water treatment system.

These plant design changes removed and/or disconnected piping and valves associated with the operation of the gaseous chlorination system. This system was removed to increase personnel safety by eliminating any possibility of an accident involving gaseous chlorine.

This modification did not result in a change to the WNP-2 Technical Specifications. This change did not result in an unreviewed safety question because the removal of gaseous chlorine from the site increases overall personnel safety and does not decrease the capabilities of the plant make-up water treatment system.

PLANT DESIGN CHANGE 84-1570

Plant Design Change 84-1570 was initiated to procure and install a single-package heat pump for the sodium hypochlorite tank storage area.

This modification installed a roof-mounted heat pump unit to serve the sodium hypochlorite tank storage area. All duct, hanger supports and accessories were also installed via this design change package.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the addition of this equipment cannot cause an unanalyzed accident.

PLANT DESIGN CHANGE 84-1623

Plant Design Change 84-1623 was initiated to install a higher capacity jockey pump for the Fire Protection System. This modification also added a still water casing to the pump to assure action of the water from adjacent pumps will not cause damage to the jockey pump.

This modification was initiated because the originally installed pump had two major failures and was determined to be inadequate.

This modification did not result in a change to WNP-2 Technical Specifications or involve an unreviewed safety question because the upgraded equipment only increases the reliability of the system; no functional system changes have been made.

PLANT DESIGN CHANGE 85-0050

Plant Design Change 85-0050 was initiated to replace the originally installed Plant Process Computer with a state-of-the-art computer system to increase reliability and system capabilities.

The originally installed plant process computer was unreliable and difficult to maintain due to the unavailability of spare parts. This design change replaced the original process computer with a state-of-the-art system designed to increase system capabilities and overall reliability.

This modification did not involve a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the overall system function has not been altered as a result of this modification.

PLANT DESIGN CHANGE 85-0080

Plant Design Change 85-0080 was initiated to provide a replacement microcomputer for the Rod Worth Minimizer System.

The plant process computer was replaced with an updated computer system that did not include the software for the Rod Worth Minimizer System. This design change package installed a microcomputer to interface with the Rod Position Instrumentation System for display and control of operator rod movements when less than the low power setpoint.

This design change did not involve a change to WNP-2 Technical Specifications or involve an unreviewed safety question because replacement system design is capable of meeting the existing Technical Specification LCO and surveillance testing requirements.

PLANT DESIGN CHANGE 85-0155

Plant Design Change 85-0155 was initiated because control air lines on originally installed dampers for the Turbine Building Exhaust Air System were freezing during winter months.

This design change replaced air-operated dampers with automatic backdraft dampers. This modification was designed for more reliable operation by installing mechanically operated dampers rather than relying on a sensing/control system.

This modification did not involve a change to WNP-2 Technical Specifications or involve an unreviewed safety question because there is no change to the function or intent of the system design.

PLANT DESIGN CHANGE 85-0273

Plant Design Change 85-0273 was initiated because the Main Steam Line pressure transmitting detectors are temperature sensitive and were influenced by ambient condition changes in the Turbine Building.

This plant modification relocated the Main Steam Line pressure transmitting detectors to an area within the Turbine Building where ambient conditions are more stable. Following the instrument relocation, leak tests, vacuum tests and visual inspections were performed in accordance with the ASME Section XI plan.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the function of the system has not changed and does not create the possibility of an unanalyzed event.

PLANT DESIGN CHANGE 85-0330

Plant Design Change 85-0330 was initiated in response to deficiencies found during the Supply System's review of the WNP-2 Appendix "R" Safe Shutdown analysis.

The majority of work associated with this design change involved the installation of fire remote transfer switches on equipment required for safe shutdown of the plant. This change ensures control power is available from either the Remote Shutdown Panel or the Alternate Remote Shutdown Panel in the event of a fire in the main control room. The other major area of work involved covering cable tray and individual cables with thermolag insulation.

This modification did not involve a change to WNP-2 Technical Specifications or involve an unreviewed safety question because the additional protection reduces the consequences of a design bases fire in the control room, thereby maintaining the plant original design bases and compliance to the intent of Appendix R.

PLANT DESIGN CHANGE 85-0519

Plant Design Change 85-0519 was initiated to provide a method for reducing erosion in the High Pressure Turbine cross-under lines.

This modification installed moisture preseparator units to the high pressure turbine extraction lines. Storage tanks with automatic level controls that discharge to the feedwater heaters were also added. Various electrical and instrumentation support was required to provide automatic control of these drain tanks.

This modification did not result in a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the design change increases the reliability of the piping and because this equipment is not considered to be safety-related.

PLANT DESIGN CHANGE 85-0676

Plant Design Change 85-0676 was initiated as a result of ongoing effects by the Supply System to mitigate effects of Intergranular Stress Corrosion Cracking (IGSCC) and in consideration of NUREG-0313, Rev. 1.

This modification replaced piping from the drain lines on loops A and B of the Reactor Recirculation Pumps with material that is not susceptible to IGSCC. This design change also removed four Pipe Whip Restraints (PWS) after an engineering analysis and walkdowns were performed verifying that no safety-related targets were within range of the affected piping.

This modification did not result in a change to the WNP-2 Technical Specifications or result in an unreviewed safety condition because pipe breaks at the locations of the removed pipe whip restraints have no safety-related targets. The other portion of this design change is a material upgrade that will enhance system reliability that has no safety concerns associated with it.

PLANT DESIGN CHANGE 86-0005

Plant Design Change 86-0005 was initiated to satisfy Regulatory Guide 1.97 requirements relative to the environmental qualifications of the wetwell level monitors.

This modification removed originally installed wetwell wide-range level monitors and replaced them with environmentally qualified instruments. The existing level recorders, as well as most interconnecting cable, were used with no modifications required.

This modification did not involve a change to the WNP-2 Technical Specifications or involve an unreviewed safety question because the upgrade of the equipment increases system reliability, thereby potentially increasing the margin of safety.

PLANT DESIGN CHANGE 86-0052

Plant Design Change 86-0052 was written in response to a General Electric recommendation to inspect certain sample probes installed at WNP-2 for signs of crevice corrosion.

This modification replaced an originally installed sample probe, RFW-SP-3, with a shorter type because when the inspection for crevice cracking was performed, it was found that the sample probe had broken off. No evidence of crevice corrosion was found on the inspected weld area.

This modification did not involve a change to the WNP-2 Technical Specifications or an unreviewed safety question because the function of the sample probe is unchanged. The stress corrosion resistance has been improved, thereby increasing overall component performance.

PLANT DESIGN CHANGE 86-0324

Plant Design Change 86-0324 was initiated because the Standby Service Water System was experiencing water hammer problems during system startup.

This modification added a keep-full pump with associated piping and controls to maintain the Standby Service Water System full under normal operating conditions. This design package also replaced originally installed system isolation valves with new valves which have better throttling and sealing characteristics.

The changes that were made reduce the potential for hydraulic transients in the piping, thereby reducing the probability of an accident associated with the system. No modification to WNP-2 Technical Specifications was made as a result of this design change.

PLANT DESIGN CHANGE 86-0569

Plant Design Change 86-0569 was initiated to address the problem of undersized circulating water pump feeders in underground duct banks.

This plant design change installed four 500-MCM cables per phase in the underground section of the circulating water pump feeder to lower all cables to within their rated temperatures. These motors previously had two 500-MCM cables per phase and experienced a failure on the feeder cable to pump 1C.

This modification did not result in a change to the WNP-2 Technical Specifications. This design change increases the conductor size for some cables in underground duct banks, thus providing an additional margin of safety by lowering the normal operating temperatures within the duct banks. As a result of this modification, an engineering analysis was performed to review the cable ampacity criteria used at WNP-2.

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 1987. Each modification was evaluated and determined not to represent an unreviewed safety question or require a change to the WNP-2 Technical Specifications.

Reactor Recirculation Mechanical Jumper

A mechanical jumper was installed between the extend and retract lines for the recirculation valve flow actuator. This jumper was installed to circulate hydraulic fluid to facilitate flushing and leak testing following refurbishment of the actuator. Upon completion of flushing and testing, the actuator was reinstalled on the flow control valve in its original configuration.

This modification did not involve an unreviewed safety question or reduce any margin of safety as defined in WNP-2 Technical Specifications because the plant was in Mode 5 and the recirculation system was not required to be in service for the time the jumper was installed.

Bypass of Low Level Trip Switch on FDR-P-21

The low level trip relay for Radioactive Floor Drains Pump 21 (FDR-P-21) was bypassed to allow a complete blowdown of a hold tank to the river. The contents of the hold tank had been processed through the condensate demineralizers, but had a high total organic carbon (TOC) level, which is not suitable for high-purity reactor water. Prior to releasing water to the river, an analysis was performed to ensure the tank content was within limits stated in the WNP-2 NPDES permit and 10CFR20, Appendix B.

The installation of this jumper did not involve an unreviewed safety question or reduce any margin of safety as defined in WNP-2 Technical Specifications. The bypassed switch function is to protect the pump from loss of suction due to a tank low level condition. The function of this switch is not related to reactor safety.

Bypass of HVAC Roll Filters Auto Advance Function

The auto advance function of the intake air filters for the Control Room, Radwaste, Turbine and Reactor Buildings was bypassed due to the variations in air flow through the buildings being greater than anticipated. The wide variations in air flow caused the dP switches to actuate the auto filter advance function unnecessarily, thereby wasting roll filters. A design change is in process to replace the dP switches with automatic timers. Until the design change is implemented, the Operations staff will continue to manually advance the filters on a routine basis.

The installation of these jumpers did not involve an unreviewed safety question or reduce any margin of safety as defined in the WNP-2 Technical Specifications because the overall function of the system has not changed from the original design and adequate compensatory measures were implemented. Periodic area temperature monitoring also verifies adequate area cooling.

Lifted Lead on Turbine Governor Valve #4

Vibration levels on a governor valve (GV) actuator has resulted in administrative limits being placed on the percent open at which GV #4 is allowed to operate. The limit was established from an empirical vibration evaluation. During normal operation, the GV is placed in the DEH "test" mode to limit valve position. Other governor valve positions and overspeed functions are not affected by this method of operation. During the monthly bypass valve testing, required by Technical Specifications, a lifted lead is relied upon to close the valve while the "test" mode of DEH is utilized to cycle all of the turbine control valves. The lead is lifted only in support of this testing. Maintenance to alleviate the GV vibration problem is scheduled for the 1988 Refueling Outage.

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.

Appendix F, Control of Combustibles

Modification - This revision to the FSAR defines all types of compressed gases stored onsite and eliminates the location of cylinder storage. All compressed gases are stored in accordance with Supply System requirements and comply with requirements as set forth in NFPA 5 and NFPA 50A.

Basis for Change - Due to an NRC inspection, modification to the Supply System program for storage of noncombustible compressed gases was made to include physical restraints at the top and bottom of gas bottles. Storage of all bottles was either modified to comply with the requirements or verified to be in compliance with the program. As a result of this effort, the FSAR was modified to reflect a list of all compressed gases stored onsite and to define the methods of storage.

Chapter 13, Conduct of Operations, Composition of Fire Brigade

Modification - The following statement was changed, "The Fire Brigade shall consist of the following personnel:" to "The Fire Brigade shall normally consist of the following personnel; however, any combination of qualified personnel meeting 10CFR50 Appendix R requirements is acceptable."

Basis for Change - This modification was made to allow greater flexibility in utilizing plant personnel for the composition of the Fire Brigade. A requirement for being a Fire Brigade Leader is to have successfully completed a training course specifically developed for that position. The limitation in the previous FSAR section, that the Fire Brigade Leader be a Shift Support Supervisor, was unnecessarily restrictive.

Appendix F, Safe Shutdown Equipment

Modification - Removes the commitment that during normal plant operation, power is removed from RHR-V-8 (suction line isolation valve).

Basis for Change - To correct an error made in a previous FSAR amendment. This section discusses the Supply System's position on resolving the high/low pressure interface system issue. This issue is currently being addressed with the NRC and an alternate solution has been proposed by the Supply System. To date, the Supply System has not de-energized RHR-V-8 or RHR-V-9 during normal operation.

Chapter 5, Residual Heat Removal (Shutdown Cooling Mode)

Modification - The paragraph describing the flow path for prewarming RHR Loop A was changed to delete RHR-V-71A from the description and to add RHR-V-70. The basic change is that rather than releasing water to radwaste from a point at pump suction (RHR-V-71A) and a point downstream of pump discharge (RHR-V-72A and RHR-V-70), only the downstream drain will be used.

Basis for Change - This modification does not reduce the amount of piping prewarmed.

Chapter 7, Engineered Safety Feature Systems

Modification - A new paragraph was written to describe the actions that must be taken to initiate the Standby Gas Treatment (SGT) System. The FSAR section deals with manual actuation capabilities of ESF Systems and SGT was not fully addressed. The clarification states that in addition to actuating the system start control switch, a control switch for a suction valve must also be actuated. This change does not involve a new method of system operation or a plant modification; it simply clarifies what actions must be taken to manually initiate SGT.

Basis for Change - To clarify what actions must be taken to manually initiate the Standby Gas Treatment System.

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 1987:

NCR 287-198 (RHR Flow Control Valve Stroke Times)

- Problem Description

  The two-year Valve Position Indication (VPI) procedures for RHR-FCV-64A, B and C were revised by means of a procedure deviation to require the valves to be operated until they were full open. Based on stem travel, the valves must be open at least 90 percent to meet the two-year VPI requirements. However, to meet this requirement, the open limit switch required adjustment to allow the valve to stroke open further to meet the requirements of the two-year VPI. As a result, this caused an increase in the stroke time.

- Corrective Action

  The NCR immediate disposition ("Use-As-Is") was to adjust the open limit switch to meet the two-year VPI requirements, measure the stroke time and revise the procedures that perform the ASME stroke time testing requirements. It should be noted that valve stroke time is not required of Technical Specification Table 3.6.3-1 because the valves do not perform a containment isolation function.

  An engineering evaluation was performed and it was determined that 1) the valves travel at the same rate and are within the 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 flow/pressure is diminished by three restriction orifices downstream.

  The 10CFR50.59 Safety Evaluation and concluded that the change does not decrease the capabilities of the RHR System or affect the minimum flow protective function.

  An FSAR Change Request was prepared to revise FSAR Table 6.2-16 to agree with the Technical Specifications.

NCR 287-340 (Reactor Recirculation-RRC-Valve Hydraulic Control Unit)

o Problem Description

Due to unacceptable oscillation in reactor recirculation flow caused by flow control valve RRC-V-60A position feedback signal problems, the hydraulic unit for the valve was shut down, rendering the valve stationary, thus preventing the recirculation loop flow runback function given a Reactor Feedwater (RFW) pump trip and Level-4 actuation.

o Corrective Action

Annunciator Response Procedure 4.603.A8-3.7 was revised by means of a procedure deviation to account for the locked-in-position condition of RRC-V-60A. The deviation provided direction to transfer both Reactor Recirculation Pumps to slow speed (15 Hz) in the event of a loss of an RFW pump and the RRC flow runback in the operable loop. The deviation presented compensatory measures evaluated as acceptable by the Supply System and General Electric.

A 10CFR50.59 Safety Evaluation was performed and concluded that the operational condition of RRC-V-60A locked in position, preventing an FCV runback in the event of an RFW pump trip and Level-4 condition, is bounded by the FSAR (Chapter 15) analysis and does not represent an Unreviewed Safety Question (USQ). The direction provided by the procedure deviation to trip both RRC pumps to 15 Hz operation compensates for the runback function in an attempt to prevent a scram. The direction merely replaces a plant reliability-based function with suitable operator actions. The flow control runback function does not perform a safety function.

NCR 298-299 (Division 1, 24-Volt Battery Replacement)

o Problem Description

During a field walkdown with a vendor representative, it was noted that the Division I, 24-Volt battery was different from the Division II battery. Further investigation revealed that the Division I battery had been replaced prior to Startup and had not been seismically qualified prior to installation.

o Corrective Action

An Engineering analysis was performed to determine required corrective actions with the following results:

- Based upon vendor-supplied information, it was determined that the battery would meet or exceed all capabilities of the originally installed battery.

- Based upon Sandia Lab seismic test results, it was determined that due to inherent ruggedness, the installed, unqualified battery was acceptable for interim use.

- Concurrent with the engineering evaluation, a 10CFR50.59 Safety Evaluation was performed and concluded that 1) the battery meets or exceeds the capabilities of the originally installed battery and, therefore, is justified for continued use, 2) the differences between this battery and the originally qualified battery are not significant and do not affect performance, and 3) to date, the battery has performed and been tested successfully to the surveillance requirements listed in the WNP-2 Technical Specifications.

- This battery will be replaced with a seismically qualified battery during the 1988 refueling outage.

NCR 287-294 (Missing Hangers on Diesel Starting Air Drain Lines)

o Problem Description

During review of a Design Change Package (DCP), it was noticed that the work package documentation was incomplete. A field walkdown revealed that two hangers for the air receiver drain lines had not been installed, although the work package showed the work as being complete.

o Corrective Action

A 10CFR50.59 Safety Evaluation was performed and concluded that the projected stress levels were not sufficient to cause a failure of the drain line; therefore, there was not a reduction in the margin of safety as previously evaluated.

Results of an Engineering Analysis allowed interim operation until the hangers could be installed. The analysis showed that the piping stresses would exceed the ASME Code requirements, but would not exceed the yield stresses of the pipe material. Therefore, although Code allowables for seismic loading were exceeded, no actual failure of the line would occur had a seismic event occurred. The hangers were fabricated and installed, restoring the system to its intended design configuration.

NCR 287-352 and 287-353 (Exceeded Thrust Loads on Safety-Related MOVs)

o Problem Description

As a result of MOVATS testing, the as-found thrust load on a number of valve operators exceeded their rated loads. The affected valves are HPCS-V-4, HPCS-V-12, RCIC-V-10, RCIC-V-13, RCIC-V-45 and RCIC-V-59. The motor operator manufacturer allows a nominal 10 percent overload with a maximum one-time overload of 250 percent. The MOVATS data taken during testing indicated the measured thrust values exceeded the motor operator design thrust limits by 20 percent to 80 percent.

o Corrective Action

A review of the as-found and as-left thrust loadings has been initiated with a consultant knowledgeable in the analysis of motor operators. Based on this initial review and evaluation of the valve operating history, the operators were approved for continued use until the 1988 refueling outage. Long-term corrective action is still being evaluated. Further analysis is being performed to formulate specific recommendations for each valve operator.

A 10CFR50.59 Safety Evaluation was performed and concluded that 1) motor operator loads expected between now and the 1988 refueling outage and the past operating history loads are within the predicted fatigue limits for these operators; therefore, the operators will perform as previously evaluated in the FSAR, 2) no new failure mechanism has been introduced and the motor operators will perform their safety function per their original design, and 3) the fatigue usage remains within the design basis acceptance limits for these operators and the margin of safety for these operators has not been reduced.

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 1987 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 8.3.64 Standby Liquid Control ATWS Preoperational Test

A plant modification was made to comply with requirements set forth in 10CFR50.62(c)(4). This test was performed to ensure that the Standby Liquid Control (SLC) system meets the minimum flow requirement of 86 gpm with an equivalent boron concentration of 13 percent. It also verified system operability following the relocation of the injection path and control circuit modifications. All acceptance criteria was met and verified to comply with the above-stated requirements.

PPM 9.5.4 Fuel Channel Measurement Acceptance Test

This procedure was prepared to implement all ASEA-ATOM acceptance criteria and tests, as well as all additional testing proposed by Supply System personnel on the new fuel channel measurement device purchased from ASEA-ATOM.

Channel deformation occurs due to neutron exposure and could potentially deform in such a manner to constrict the free movement of the control rods. Deformation could also impact the LPRM and result in apparent flux tilts and erroneous flux profile indications. With the fuel channel measurement device, this deformation can be measured and acceptance criteria established to eliminate the grossly deformed channels and re-use the acceptable channels.

This procedure verified the correct installation and function of the fuel channel measurement device and was determined to be acceptable for use.

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 1987, 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.

During September, a formal 10CFR50.59 Evaluation procedure was issued and, in accordance with that process, the following procedural revision was identified as being a change to a procedural commitment as described in the FSAR:

o PPM 1.3.03, "Fire Protection Program Training"

The purpose of this procedure is to establish the basic guidelines for Fire Brigade, Fire Watch and Fire Fighting which are consistent with regulatory commitments, recognized industry practices and Supply System programs. The procedure applies to the training of personnel for these programs.

The procedure was revised to provide (a) consistency with 10CFR50, Appendix R, requirements and (b) flexibility regarding assignments made to the Fire Brigade.

As a result of a 10CFR50.59 evaluation, it was determined that the revision constituted a change to a procedural commitment as described in the FSAR. Accordingly, an SAR Change Notice (SCN87-061) was prepared and will be sent to the NRC.

The following is a discussion of significant plant procedure changes and development during 1987:

1. PPM 1.3.5, "Reactor Trip and Recovery"

The purpose of this procedure is to define the process for determining the causes of a reactor trip, provide guidelines for reactor restart and document the plant response to the trip.

The procedure was revised as a result of a thorough review of plant operations following the five unplanned scrams discussed previously in the Introduction Section of this report.

The following improvements were included in the revision:

- The follow-up review committee will be assembled immediately following any reactor trip unless otherwise directed by the Plant Manager.
- When personnel performance is a contributing factor to an event, a peer review committee will be formed to evaluate the decision process.
- Predetermined experts will be relied upon for cause investigation in support of the restart decision process. It should be noted that this aspect is currently effectively performed and the change is to document who the resources are and facilitate management of the group.
- If the cause(s) of a reactor trip are indeterminate, a surveillance method will be developed, if possible, to monitor the component/system response following restart.
- Independent verification will be required of significant conclusions relied upon in the restart decision process and root cause assessment.

2. PPM 1.3.43, "10CFR50.59 Evaluation Process"

The purpose of this procedure is to provide guidance for performing a 10CFR50.59 evaluation and identify the plant procedures subject to the 10CFR50.59 evaluation process.

The procedure was developed to formalize the 10CFR50.59 evaluation process at WNP-2. This procedure identifies the fractional parts of the evaluation process and establishes the method for evaluating changes to structures, systems, components, procedures and proposed tests or experiments according to the requirements of 10CFR50.59. It is not the intent of the procedure to, in any way, limit required safety evaluations to only those proposed changes described in 10CFR50.59.

3. Instrument Program Procedures

- PPM 1.4.3, "Revision of Master Data Sheets and Setpoints"

  The purpose of this procedure is to establish the responsibilities for the control of all plant setpoints and to provide a procedure for the preparation, review and approval of setpoint changes or other general Master Data Sheet changes. This also includes the control of calibration tolerances and Administrative Limits for instruments addressed in the Technical Specifications.

- PPM 1.4.4, "Plant Instrument Design Documentation"

  The purpose of this procedure is to provide instructions for the preparation and control of Instrument Master Data Sheets.

- PPM 1.4.12, "Instrument Setpoint Calculations"

  The purpose of this procedure is to provide instructions for the preparation or review of instrument setpoint calculations.

These three procedures, which govern the Instrument Program at WNP-2, were revised to assure that setpoints are set equal or conservative to the Technical Specification Limits. The revisions were the result of an NRC Notice of Violation (NOV87-26) where it was identified that Main Steam Line Tunnel Temperature-High setpoints for channels "A", "B", "C" and "D" were set to a value (156°F) higher than allowed (150°F) by the Technical Specifications.

The Supply System acknowledged the validity of the violation in that the Main Steam Tunnel Leak Detection instrumentation was calibrated to a value higher than the Technical Specification Trip Setpoint without prior NRC approval. However, it should be noted that the decision to calibrate the MSL Tunnel instruments to the allowable limit was based on our interpretation of the Technical Specification Bases (Section 3/4.3.2). It is stated in the Bases that, "Operation with a trip set less conservative than its trip setpoint but within the specified allowable value is acceptable on the basis that the difference between each trip setpoint and the allowable value is equal to or less than the drift allowance assumed for each trip in the safety analysis."

This interpretation was considered to be a programmatic concern and, as a result, the procedures were revised accordingly.

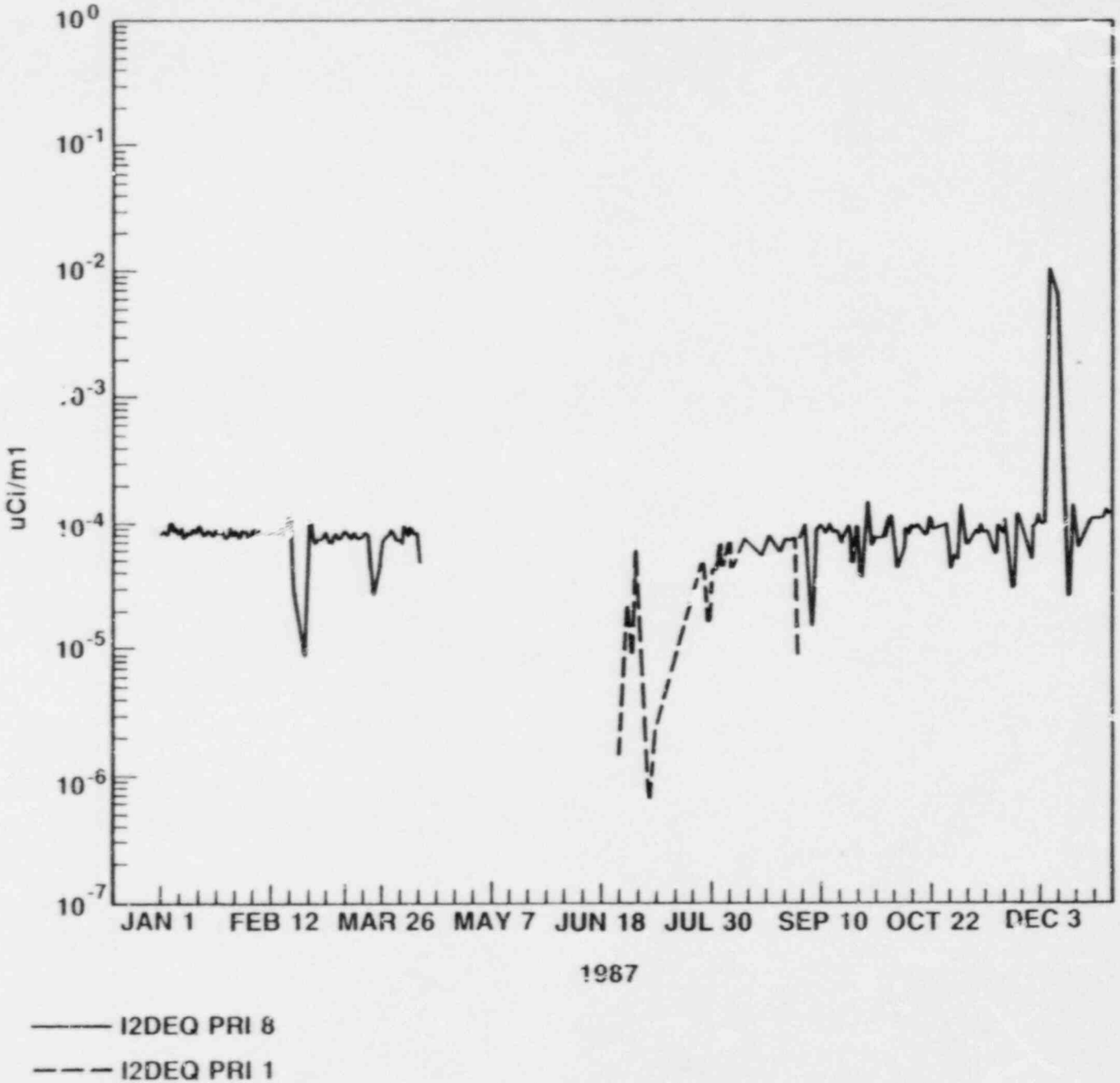
2.9 REACTOR COOLANT ACTIVITY CUMULATIVE IODINE LEVELS

This section contains information relative to reactor coolant cumulative iodine levels and iodine spikes. The specific activity of the primary coolant was significantly less than the limits of (a) less than or equal to 0.2 microcuries per gram dose equivalent I-131, and (b) less than or equal to 100/E microcuries per gram as set forth in WNP-2 Technical Specifications.

Following a scheduled reactor shutdown, water sample analysis confirmed previous indications of fuel failure. The iodine spike noted on December 6, reflects increased iodine concentrations due to the fuel pin failure.

A graph showing cumulative iodine dose equivalent for the calendar year 1987 is provided for reference and completeness. This information is provided in accordance with WNP-2 Technical Specifications.

WNP-2
DOSE EQUIVALENT IODINE



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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 25, 1988

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 1987. Should you have any questions or comments please contact M. R. Wuestefeld, WNP-2 Plant Engineering Supervisor, Reactor Systems.

Very truly yours,

C. M. Powers
Plant Manager

CMP:MRW:TRW

Attachments

cc: Dottie Sherman, ANI
Document Control Desk, NRC

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