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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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

February 28, 1992
G02-92-054

Docket No. 50-397

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: **NUCLEAR PLANT WNP-2, ANNUAL OPERATING REPORT 1991**

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

2) WNP-2 Technical Specifications, 6.9.1.4 and 6.9.1.5

3) Regulatory Guide 1.16, Reporting of Operation Information Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Operating Report for calendar year 1991. Should you have any questions or comments, please contact G. L. Gelhaus, WNP-2 Assistant Plant Technical Manager.

Very truly yours,

J W Baker

J. W. Baker
WNP-2 Plant Manager

Enclosure

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

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ML14350A147

Forwards Response to NRC Recommendations Contained in SER of Station Blackout Analysis. Dated March 6, 1992.

431-97-0141 box #8

ML14350A149

Suppls 920306 response to NRC SER re: station blackout analysis, consisting of listing of station blackout containment isolation valves. Dated May 15, 1992.

431-97-0141 box #8

WASHINGTON NUCLEAR PLANT No. 2

ANNUAL OPERATING REPORT

FOR 1991

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 1991 Annual Operating Report of Washington Public Power Supply System Plant Number 2 (WNP-2) is submitted pursuant to the requirements of Federal Regulations and Facility Operating License NPF-21. Plant WNP-2 is a 3323 MWt, BWR-5, which began commercial operation on December 13, 1984.

WNP-2 started out 1991 by setting a record for single month generation. The January output was 798 million kilowatt-hours electric with a 97.9% capacity factor. February and March generation were also records for those months, respectively. The Plant continued to operate for 101 consecutive days until a planned outage was taken on March 21 to clean transformer insulators. The reactor was not shutdown, but the generator was taken off-line. The Plant was returned to power after a 12 hour outage and operated until April 13 when indication of wear particles in a diesel generator lube oil sample led Plant management to declare the diesel inoperable and to enter the scheduled refueling outage R6.

In March several operators and operations crews failed their NRC requalification examinations. Additional failures also occurred in June. The Supply System performed a major upgrade of the Emergency Operating Procedures (EOPs) to conform to the latest guidance available. Upgraded operator training for EOPs, particularly the addition of more challenging simulator scenarios, was implemented. These programs required an immense effort from a broad spectrum of Company personnel. This resulted in EOPs which provide clearer guidance, and operators who have experience dealing with demanding accident sequences. The negative consequences of these improvements, however, was a maintenance and refueling outage that was administratively extended to over five months in duration to provide adequate time for the improvements to be made.

In the remaining months of the year following the outage the Plant experienced three shutdowns and two power reductions due to increasing reactor coolant conductivity as a result of condenser leakage. The Plant also scrammed in November due to a defective capacitor in the Feedwater Level Control system.

In addition to the activities described above, during 1991 there were several other examples of major accomplishments which required significant effort on the part of Supply System personnel to complete. The following is a summary of those efforts.

The sixth refueling outage was successfully completed on schedule and within budget. Significant activities for the outage included:

- Performance of an Integrated Leakage Rate Test of the Primary Containment.

- Replacement of the rubber seal between one low-pressure turbine and the condenser box to minimize air leakage into the condenser.

- Completion of the Main Turbine Governor Valve Optimization Program which resulted in a 10 MW improvement.

- Four Main Steam Bypass Valves rebuilt for improved reliability.

- Inspection of all five of the Emergency Diesel Generator engines.

- Refurbishment of 90 MOV actuators, baseline testing of 34 MOVs, and differencial pressure testing of 22 MOVs

- Inspection of one of three Low-Pressure Turbine Rotors. Non-destructive examination of the rotor confirmed crack indications and twenty one blades were replaced.

- Removal of spent fuel assemblies and refueling the reactor. The refueling activity included replacing 120 fuel assemblies using a fuel shuffle scheme.

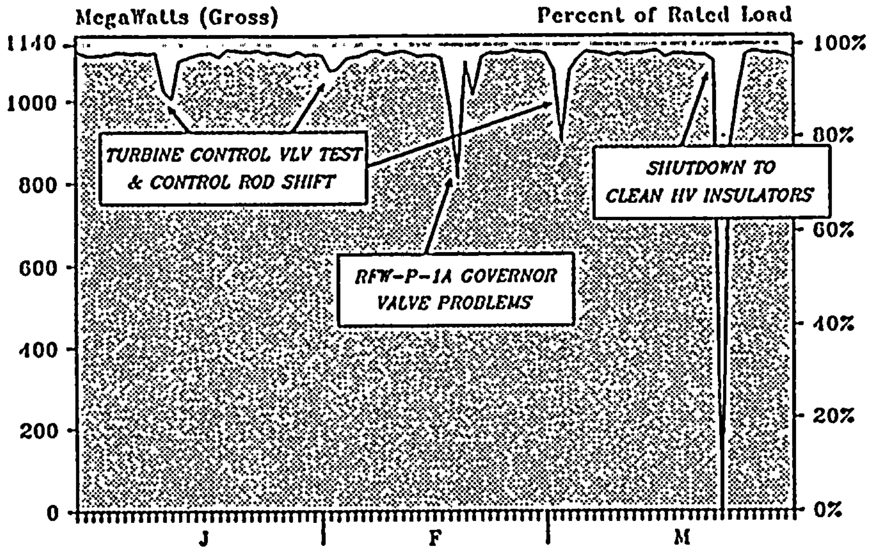
In 1991 total radiation exposure at the Plant was 386.8 man-rem versus the 1990 total of 535 man-rem. This compared extremely well to a Plant goal of 480 man-rem, a positive indication of the ALARA commitment at WNP-2.

The 1991 capacity factors, based upon net electrical energy output, are listed in the following table.

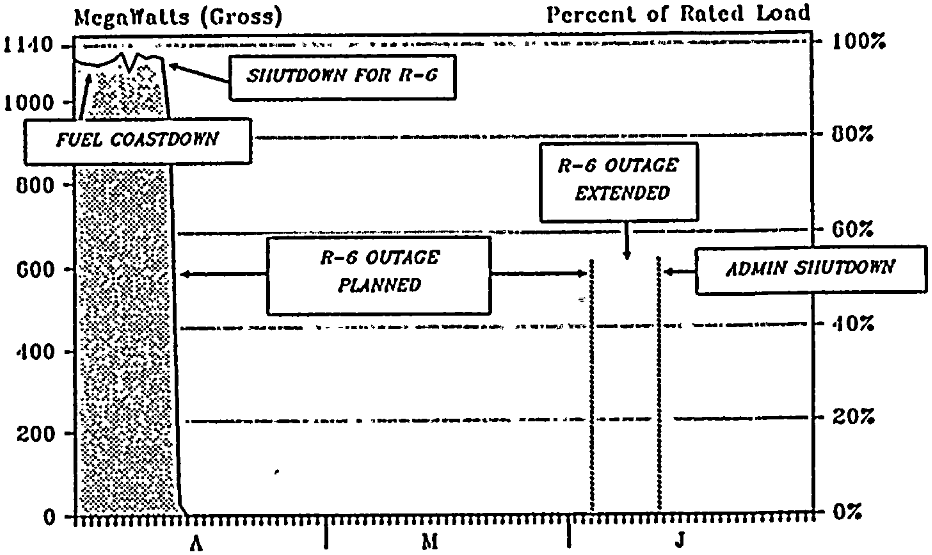
Month	Capacity Factor
January	97.9
February	96.5
March	93.5
April *	37.8
May	0
June **	0
July	0
August	0
September	0
October	72.9
November	60.6
December	79.7
Overall	44.3

* Started Maintenance/Refueling Outage
** Ended Maintenance/Refueling Outage, Entered Administrative Outage

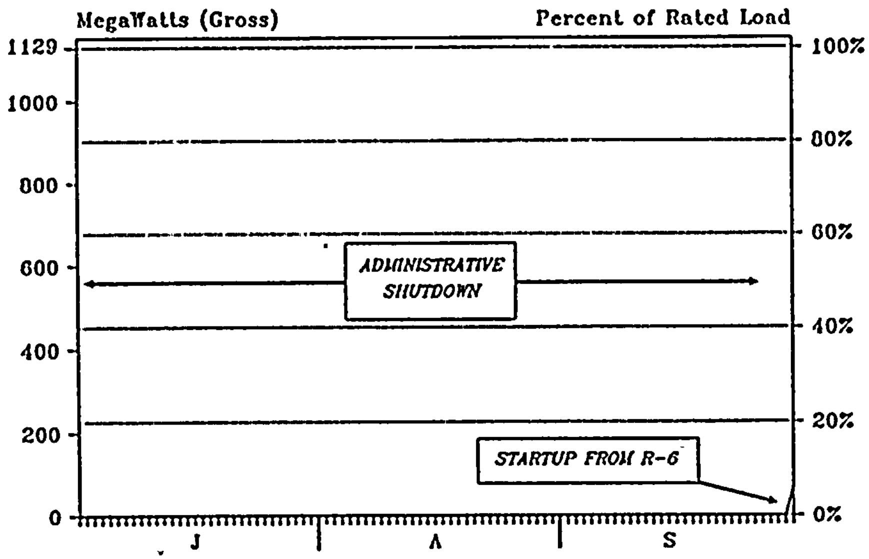
1.1 WNP-2 LOAD PROFILE FOR 1991



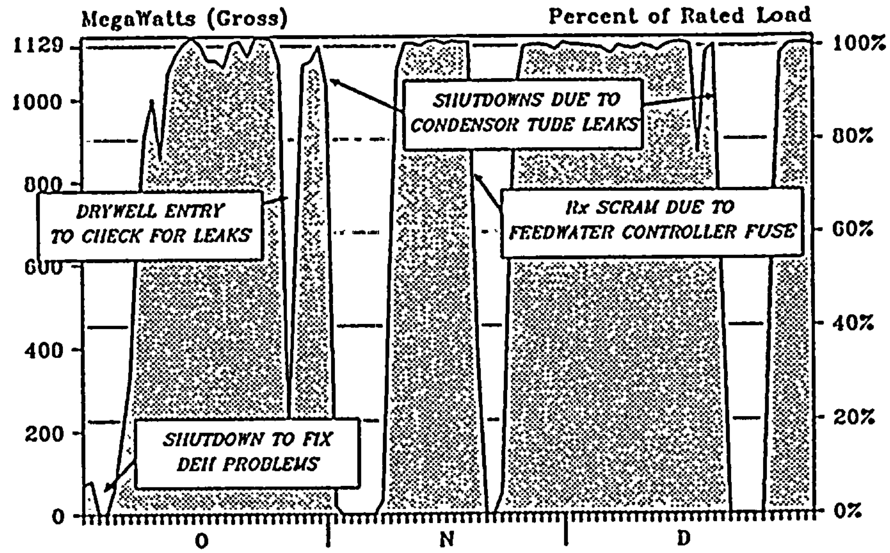
1st QUARTER 1991



2nd QUARTER 1991



3rd QUARTER 1991

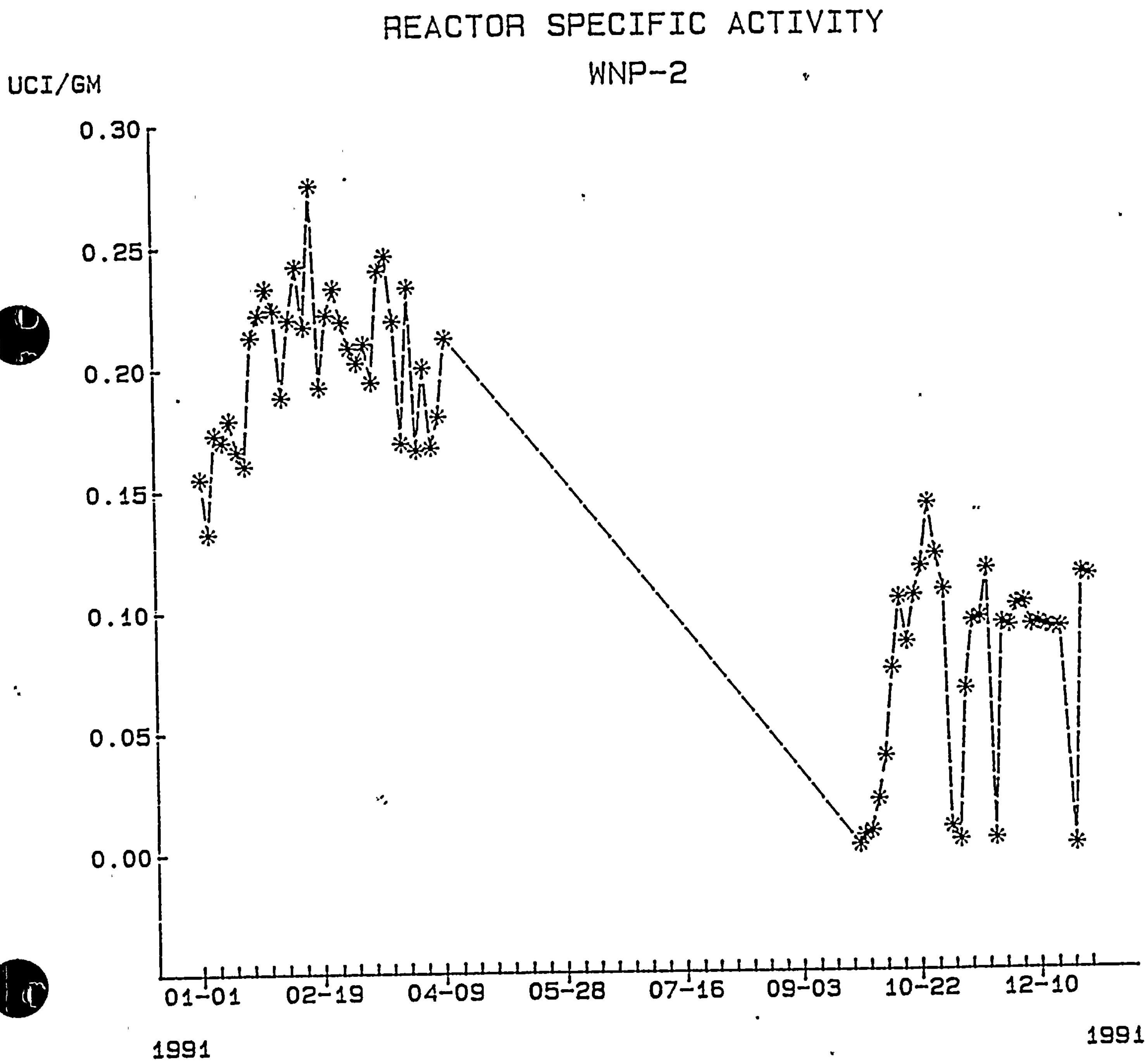


4th QUARTER 1991

1.2 REACTOR COOLANT SPECIFIC ACTIVITY LEVELS

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

The specific activity of the primary coolant was significantly less than 0.2 microcuries per gram dose equivalent I-131 as set forth in WNP-2 Technical Specification LCO 3/4.4.5. In addition, as shown below, 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 100/E-bar microcuries per gram.



REACTOR SPECIFIC ACTIVITY 100/E-BAR UCI/GM

2.0 REPORTS

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

2.1 ANNUAL PERSONNEL EXPOSURE AND MONITORING REPORT

The information provided in this section of the report is required by the WNP-2 Technical Specifications, Section 6.9.1.5.a, and Regulatory Guide 1.16, Revision 4. These values are estimated doses for the listed activities based on pocket dosimetry readings.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT /1.16 APPENDIX A
NUCLEAR PLANT NO. 2 REPORT FOR CALENDAR 1991

	NUMBER RECIEVING OVER 100 MREM			TOTAL MAN REM		
	STATION EMPLOYEE	UTILITY EMPLOYEE	CONTRACTOR AND OTHERS	STATION EMPLOYEE	UTILITY EMPLOYEE	CONTRACTOR AND OTHERS
OPERATION AND SURVIELLANCE						
MAINTENANCE PERSONNEL	38	1	10	17.1	0.3	3.7
OPERATING PERSONNEL	29	0	0	28.3	0.0	0.0
HEALTH PHYSICS PERSONNE	31	0	25	18.6	0.0	15.8
SUPERVISORY PERSONNEL	8	0	0	2.2	0.0	0.0
ENGINEERING PERSONNEL	5	2	2	1.7	0.8	0.3
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	144	1	149	88.5	0.1	79.5
OPERATING PERSONNEL	8	0	0	9.3	0.0	0.0
HEALTH PHYSICS PERSONNE	13	0	14	15.2	0.0	12.5
SUPERVISORY PERSONNEL	8	2	1	2.4	0.6	0.5
ENGINEERING PERSONNEL	16	18	18	6.9	8.1	6.5
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	2	0	1	0.6	0.0	1.2
OPERATING PERSONNEL	0	0	0	0.2	0.0	0.0
HEALTH PHYSICS PERSONNE	1	0	0	0.7	0.0	1.1
SUPERVISORY PERSONNEL	0	0	0	0.0	0.2	0.0
ENGINEERING PERSONNEL	1	0	1	0.6	0.2	0.6
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	15	0	16	3.3	0.0	8.3
OPERATING PERSONNEL	0	0	0	0.1	0.0	0.0
HEALTH PHYSICS PERSONNE	1	0	1	0.5	0.0	1.1
SUPERVISORY PERSONNEL	0	0	0	0.1	0.0	0.0
ENGINEERING PERSONNEL	2	1	2	0.5	0.4	0.5
WASTE PROCESSING						
MAINTENANCE PERSONNEL	1	0	1	0.9	0.0	0.1
OPERATING PERSONNEL	0	0	0	0.1	0.0	0.0
HEALTH PHYSICS PERSONNE	0	0	1	0.1	0.0	0.2
SUPERVISORY PERSONNEL	0	0	0	0.1	0.0	0.0
ENGINEERING PERSONNEL	0	0	0	0.0	0.0	0.0
REFUELING						
MAINTENANCE PERSONNEL	24	0	19	28.1	0.0	5.7
OPERATING PERSONNEL	4	0	0	4.1	0.0	0.0
HEALTH PHYSICS PERSONNE	1	0	9	1.5	0.0	4.1
SUPERVISORY PERSONNEL	4	0	2	1.3	0.0	0.5
ENGINEERING PERSONNEL	3	2	8	0.8	1.4	1.7
TOTAL						
MAINTENANCE PERSONNEL	224	2	196	138.5	0.4	98.4
OPERATING PERSONNEL	41	0	0	42.0	0.0	0.0
HEALTH PHYSICS PERSONNE	47	0	50	36.5	0.0	34.7
SUPERVISORY PERSONNEL	20	2	3	6.1	0.8	1.0
ENGINEERING PERSONNEL	27	23	31	10.6	11.0	9.7
GRAND TOTAL	359	27	280	233.7	12.1	143.8

VALUES ARE BASED ON SELF READING DOSIMETER DATA.
MAN REM VALUES ARE ROUNDED TO NEAREST 0.1 MAN REM.

2.2 MAIN STEAM LINE SAFETY/RELIEF VALVE CHALLENGES

This section contains information concerning main steam line safety/relief valve (SRV) challenges for calendar year 1991 in accordance with the requirements of WNP-2 Technical Specification paragraph 6.9.1.5(b).

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)*	REASON FOR ACTUATION (CODE)*	PRIOR PLANT CONDITIONS (CODE)*	POWER LEVEL %	ASSOCIATED LER
03/22/91	MS-RV-2A	B	C	E	35	--
04/13/91	MS-RV-1B	C	C	C	0	--
04/13/91	MS-RV-4B	C	C	D	0	--
04/13/91	MS-RV-4A	C	C	D	0	--
04/03/91	MS-RV-3A	C	C	D	0	--
04/13/91	MS-RV-3C	C	C	D	0	--
09/28/91	MS-RV-2C	C	C	C	3	--
09/28/91	MS-RV-5C	C	C	C	3	--
09/28/91	MS-RV-4D	C	C	C	3	--
09/28/91	MS-RV-1B	C	C	C	3	--
09/28/91	MS-RV-2B	C	C	C	3	--
09/28/91	MS-RV-5B	C	C	C	3	--
[illegible]/91	MS-RV-1A	B	C	C	11	--
09/30/91	MS-RV-2A	B	C	C	12	--
09/30/91	MS-RV-3A	B	C	C	11	--
09/30/91	MS-RV-4A	B	C	C	12	--
09/30/91	MS-RV-1B	B	C	C	11	--
09/30/91	MS-RV-2B	B	C	C	12	--
09/30/91	MS-RV-3B	B	C	C	12	--
09/30/91	MS-RV-4B	B	C	C	12	--
09/30/91	MS-RV-5B	B	C	C	12	--
09/30/91	MS-RV-1C	B	C	C	11	--
09/30/91	MS-RV-2C	B	C	C	12	--
09/30/91	MS-RV-3C	B	C	C	12	--
09/30/91	MS-RV-4C	B	C	C	11	--

DATE	COMPONENT ID	TYPE OF ACTUATION (CODE)*	REASON FOR ACTUATION (CODE)*	PRIOR PLANT CONDITIONS (CODE)*	POWER LEVEL %	ASSOCIATED LER
09/30/91	MS-RV-5C	B	C	C	12	--
09/30/91	MS-RV-1D	B	C	C	12	--
09/30/91	MS-RV-2D	B	C	C	12	--
09/30/91	MS-RV-3D	B	C	C	11	--
09/30/91	MS-RV-4D	B	C	C	11	--
10/04/91	MS-RV-3B	B	C	C	17	--
12/25/91	MS-RV-4D	B	C	E	14	--
11/19/91	MS-RV-5B	B	E	G	0	--
11/19/91	MS-RV-3D	B	E	G	0	--

<u>NOTES</u>:

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

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

CODES:

<u>Type of Actuation</u>

A. Automatic
B. Remote Manual
C. Spring

<u>Plant Condition</u>

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

<u>Reason for Actuation</u>

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

2.3 SUMMARY OF PLANT OPERATION

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

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE (CODE)	SHUT DOWN (CODE)	LER NUMBER	SYSTEM	COMPONENT	CAUSE & ACTION TO PREVENT RECURRENCE
3/21/91	S	12.7	B	1	--	EB	ELCON	Generator was removed from service for cleaning of high voltage insulators due to buildup of chemical deposits from cooling tower drift. After completion of cleaning, the plant was returned to service.
4/13/91	S	1435	A	1	91-006	RC	FUEL XX	Plant was shut down as scheduled for refueling outage R-6. The outage started early due to a diesel generator inoperability.
6/12/91	F	2662.7	E	9	--	--	--	Remained shutdown to upgrade plant Emergency Operating Procedures and provide additional licensed operator training.
10/1/91	S	71.8	B	1	--	HA	PPIX	Plant was manually shutdown to correct turbine DEH piping and to repair steam leaks.
10/5/91	S	2.2	B	1	--	HA	MECFUN	Generator was removed from grid to perform overspeed testing of turbine-generator.
10/26/91	S	8.2	B	1	--	CB	VALVEX	Generator was removed from grid and reactor power reduced to 10% thermal power for drywell entry in search of leaks. Corrected leakage through valves RRC-V-51A/52A and returned to full power.
11/1/91	F	160.0	A	1	--	HC	HTEXCH	Plant was shutdown due to increasing conductivity of the reactor coolant. Condenser in leakage was repaired and plant returned to service.

2.3 SUMMARY OF PLANT OPERATION (continued)

DATE	OUTAGE TYPE	GENERATOR OFF-LINE HOURS	CAUSE (CODE)	SHUT DOWN (CODE)	LER NUMBER	SYSTEM	COMPONENT	CAUSE & ACTION TO PREVENT RECURRENCE
11/19/91	F	78.4	A	3	91-032	CH	INSTRU	Auto scram from 100% power due to turbine trip on reactor high level. RPV high level was caused by RFW drive turbines ramping up due to a blown fuse on feedwater controller. The problem was corrected and plant returned to service.
12/17/91	F	0	A	5	--	--	--	A power reduction to 63% thermal power was necessitated by high and rising conductivity of the reactor coolant. After one circulating water pump was removed from service, the conductivity returned to normal and the plant was returned to full power.
12/20/91	F	132.2	A	2	91-035	HC	HTEXCH	The plant was shutdown because of high and rising conductivity of the reactor coolant. A condenser tube leak was repaired and other minor maintenance was performed in drywell prior to returning plant to service.

CAUSE CODE	TOTAL FOR 1991	TOTAL GENERATOR OFF-LINE HOURS
A	5	370.60
B	4	94.90
C	1	1435
D	0	0
E	1	2662.7
	TOTAL	467.50

2.3 SUMMARY OF PLANT OPERATION (Continued)

SUMMARY OF CODES

OUTAGE TYPE	CAUSE CODE	SHUTDOWN METHOD	SYSTEM CODE	SYSTEM DESCRIPTION
F - Forced	A - Equipment Failure	1 - Manual	CB	Containment and associated components
S - Scheduled	B - Maintenance or Test	2 - Manual Scram	CH	Feedwater Systems & Controls
	C - Refueling	3 - Auto Scram	EB	AC Onsite Power Systems & Controls
	D - Regulatory Restrict.	4 - Continued	HA	Turbine Generator & Controls
	E - External Cause	5 - Reduced Load	HC	Main Condenser Systems & Controls
	F - Administration	9 - Other		
	G - Personnel Error			
	H - Other			

SUMMARY OF COMPONENT CODES

COMPONENT TYPE/CODE	COMPONENT TYPE INCLUDES:
Electrical Conductors (ELECON)	Bus Cable Wire
Fuel Elements (FUELXX)	
Heat Exchangers (HTEXCH)	Condensers Coolers Evaporators Regenerative Heat Exchangers Steam Generators Fan Coil Units
Instrumentation and Controls (INSTRU)	Controllers Sensors/Detectors/Elements Indicators Differentials Integrators (Totalizers) Power Supplies Recorders Switches Transmitters Computation Modules
Pipes, Fittings (PIPEXX)	Pipes Fittings
Valves (VALVEX)	Valves Dampers

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

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

SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT

Component ID	Failure Date	Description
CRA-FN-1B1	07/07/91	OPERATIONS PERSONNEL FOUND LINE FUSES BLOWN ON ALL 3 PHASES OF CONTAINMENT RETURN AIR FAN CIRCUITRY (CRA-M/FN-1B1) WHEN ATTEMPTING TO RETURN THE COMPONENT TO SERVICE FOLLOWING MAINTENANCE. (A NEW MOTOR HAD BEEN INSTALLED). THIS FAILURE HAD NO PLANT/SYSTEM EFFECT AS THE COMPONENT HAD BEEN OUT OF SERVICE TO FACILITATE REFUELING. TROUBLESHOOTING OF CIRCUIT FOUND NO ELECTRICAL PROBLEMS. VISUAL INSPECTION OF THE FAN/MOTOR ASSEMBLY INDICATED NO VISIBLE RUBBING. FAN PITCH WAS 27 DEGREES. PITCH SHOULD HAVE BEEN 24 DEGREES. CAUSE OF THE FUSE FAILURE IS BELIEVED TO BE AN OVERLOAD CONDITION ON THE MOTOR DUE TO MISADJUSTMENT OF THE FAN BLADE PITCH DURING FAN INSTALLATION FOLLOWING PREVIOUS REPAIR. THE FAN BLADES WERE REPITCHED TO 24 DEGREES.
CRD-HCU-2247	11/03/91	DURING TIMING ADJUSTMENT OF THE DIRECTIONAL CONTROL VALVES FOR THE CONTROL ROD DRIVE HYDRAULIC CONTROL UNIT 2247, IT WAS NOTED THAT THE WITHDRAW TIME REQUIRED TO OBTAIN A SINGLE NOTCH WITHDRAW WAS 75.0 SECONDS . THIS DEGRADES THE FUNCTION OF THE CONTROL ROD DRIVE HYDRAULIC CONTROL UNIT BUT DOES NOT AFFECT THE PLANT. DETERIORATION OF THE DIRECTIONAL CONTROL VALVE INTERNALS DUE TO AGING AND WEAR. REPLACED THE DIRECTIONAL CONTROL VALVE. ALL RESPONSE

Component ID	Failure Date	Description
		TIME TESTING RESULTS SATISFACTORY.
CRD-HCU-5427	10/08/91	DURING PLANT STARTUP FOLLOWING THE ANNUAL REFUELING OUTAGE R-6, THE CONTROL ROD WAS OBSERVED TO BE TRIPLE NOTCHING IN THE WITHDRAW DIRECTION DURING NORMAL ROD PATTERN ADJUSTMENTS BY REACTOR OPERATORS PERFORMING THE ADJUSTMENTS. THE FAILURE WAS THE RESULT OF DETERIORATION OF THE INTERNALS OF THE DIRECTIONAL CONTROL VALVES DUE TO AGING AND WEAR. ALL 4 DIRECTIONAL CONTROL VALVES WERE REPLACED. ALL SUBSEQUENT RESPONSE TESTING WAS SATISFACTORY.
DCW-HX-1A1	08/14/91	DURING PERFORMANCE OF THE DIESEL GENERATOR "A" OPERABILITY TEST FOLLOWING ANNUAL MAINTENANCE, THE TECHNICAL STAFF ENGINEER NOTED THAT THE ENGINE "1A1" TEMPERATURES WERE HIGHER THAN NORMAL. THE TEST WAS TERMINATED AND THE SYSTEM PLACED OUT OF SERVICE. THIS HAD NO EFFECT ON THE SYSTEM WHICH WAS IN TEST WHILE THE PLANT WAS SHUTDOWN FOR ANNUAL REFUELING. THE CAUSE OF THE FAILURE WAS DIRTY TUBES INTERNAL TO THE HEAT EXCHANGER ON THE SERVICE WATER SIDE. THE TUBES WERE ROUTED OUT, THE TUBE SHEET TO CHANNEL GASKET WAS REPLACED, AND THE WATER BOX DIVIDER PLATE EDGES WERE WELD REPAIRED WHERE THEY SEAL AGAINST THE WATER BOX COVER PLATE AND HEAT EXCHANGER TUBE SHEET. PERFORMANCE OF THE OPERABILITY TEST WAS SATISFACTORY AND THE DIESEL WAS RETURNED TO SERVICE.
DEH-HPU-1	10/02/91	THE MAIN TURBINE HIGH PRESSURE PISTON TYPE ACCUMULATOR (DEH-TK-1B) WAS OBSERVED TO HAVE AN OIL LEAK. THIS WAS OBSERVED BY INDIVIDUALS PERFORMING MAINTENANCE ON AN UNRELATED COMPONENT. THERE WAS NO EFFECT THE PLANT WHICH WAS SHUTDOWN. THE DEH (DIGITAL ELECTRO-HYDRAULIC)

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (cont.)

Component ID	Failure Date	Description
		SYSTEM WAS RENDERED INOPERABLE BY THE LEAK. WEAR OF THE ACCUMULATOR BOSS O-RING. THE O-RING WAS REPLACED.
DG-ENG-1C	06/10/91	DURING REVIEW OF THE VIBRATION DATA OBTAINED AS PART OF THE DIESEL GENERATOR OPERABILITY TESTING FOLLOWING THE ANNUAL MAINTENANCE, IT WAS NOTED THAT THE ENGINE EXHIBITED HIGHER THAN NORMAL VIBRATION. THE DIESEL GENERATOR WAS SHUTDOWN AND PLACED OUT OF SERVICE AS A RESULT OF THE HIGH VIBRATION. THIS HAD NO EFFECT ON THE PLANT WHICH WAS IN IT'S ANNUAL REFUELING OUTAGE, BUT DID REQUIRE THAT THE HIGH PRESSURE CORE SPRAY DIESEL GENERATOR AND HIGH PRESSURE CORE SPRAY SYSTEM BOTH BE DECLARED INOPERABLE IN ACCORDANCE WITH PLANT PROCEDURES. INVESTIGATION REVEALED THAT THE GENERATOR HAD NOT BEEN PROPERLY ALIGNED TO THE ENGINE FOLLOWING MAINTENANCE. THE ENGINE WAS REALIGNED TO THE GENERATOR AND THE COUPLING WAS DISASSEMBLED AND INSPECTED. THE COUPLING SHOWS SIGNS OF WEAR AND WILL BE REPLACED AS SOON AS REPLACEMENT PARTS ARE PURCHASED BUT DOES NOT PROHIBIT THE DIESEL GENERATOR FROM OPERATING ON DEMAND.
DLO-P-3B1	07/06/91	DURING ROUTINE ANALYSIS OF VIBRATION DATA BY CONDITION MONITORING ENGINEER, IT WAS REVEALED THAT THE DIESEL ENGINE LUBE OIL PIPING WAS VIBRATING EXCESSIVELY. FIELD OBSERVATIONS VERIFIED THE VIBRATION SPECTRUM DATA. THIS HAD NO PLANT EFFECT AS THE UNIT WAS SHUTDOWN FOR REFUEL BUT DID DEGRADE THE AVAILABILITY OF DLO-P-3B1 TO PROVIDE LUBE OIL TO THE DIESEL ENGINE. THE CAUSE OF THE HIGH VIBRATION WAS DISINTEGRATION OF THE RUBBER BISCUIT INSIDE THE COUPLING (PIECE PART) BETWEEN THE PUMP AND MOTOR. EXACT CAUSE OF FAILURE IS UNKNOWN BUT

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (cont.)

Component ID	Failure Date	Description
		SUSPECT NORMAL WEAR. REPLACED THE COUPLING AND RUBBER BISCUIT BETWEEN THE PUMP/MOTOR ASSEMBLY FOR DLO-P-3B1. THERE WAS NO SYSTEM OR PLANT EFFECT SINCE THE PUMP IS NOT REQUIRED FOR OPERABILITY.
DLO-P-3B2	07/10/91	WITH THE DIVISION 2 DIESEL GENERATOR AND SUBSYSTEMS OUT OF SERVICE FOR SCHEDULED MAINTENANCE, THE OPERATOR ON SHIFT TOUR NOTED THE COUPLING ON THE CIRCULATING LUBE OIL PUMP (3B2) WAS BROKEN. THERE WAS NO SYSTEM OR PLANT EFFECT SINCE THE PUMP IS NOT REQUIRED FOR OPERABILITY. APPARENTLY THE PUMP WAS MISALIGNED ON A PREVIOUS INSTALLATION CAUSING EXCESS STRESS ON THE COUPLING LEADING TO BREAKAGE. THE PUMP WOULD NOT HAVE RUN, BUT THERE IS A STANDBY DC POWERED PUMP AVAILABLE FOR CIRCULATING OIL, SO NO EFFECT ON PLANT OR SYSTEM. THE COUPLING WAS REPLACED WITH SAME TYPE NEW COUPLING. THE PUMP FUNCTION WAS TESTED ACCEPTABLE.
DMA-M-FN/21	08/18/91	DURING PERFORMANCE OF THE SEMI-ANNUAL DIESEL GENERATOR "B" OPERABILITY TEST, THE DIESEL MIXED AIR (DMA) FAN MOTOR TRIPPED OFF SEVERAL TIMES DUE TO THE "B" PHASE THERMAL OVERLOAD HEATER OPENING. THE BUS VOLTAGE AT THE TIME WAS 503 VOLTS WHICH WAS NOT SUFFICIENTLY HIGH ENOUGH TO REQUIRE THE HEATERS TO OPEN. THIS HAD NO EFFECT (THE UNIT WAS SHUTDOWN FOR REFUEL) BUT DID REQUIRE THAT THE DIESEL GENERATOR BE RUN IN A CONDITION WITH THE POTENTIAL TO EXCEED AREA TEMPERATURE LIMITS AND REQUIRE THE DIESEL GENERATOR TO BE SHUTDOWN AND PLACED OUT OF SERVICE. THE CAUSE OF THE FAILURE WAS PREMATURE OPENING OF THE "B" PHASE THERMAL OVERLOAD HEATER IF THE MOTOR CONTROL CIRCUIT. THE THERMAL OVERLOAD RELAY AND HEATERS WERE REPLACED.

Component ID	Failure Date	Description
		ALL TESTING WAS SATISFACTORY.
HPCS-MO-10	07/25/91	DURING PERFORMANCE OF VALVE LINEUPS FOR THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM PRIOR TO PLACING THE SYSTEM IN SERVICE FOLLOWING MAINTENANCE, THE 10-INCH MOTOR OPERATED HPCS RETURN TO THE CONDENSATE STORAGE TANK (HPCS-MO-10) VALVE WOULD NOT OPEN . THIS HAD NO PLANT EFFECT SINCE THE PLANT WAS SHUTDOWN FOR REFUELING BUT DID RESULT IN THE LOSS OF THAT SPECIFIC FLOW PATH. THE CAUSE OF FAILURE WAS AN OPEN COIL IN THE MOTOR STARTER (HPCS-42-4A2E) FOR HPCS-MO-10 WHICH IS CONSIDERED A PIECE PART OF THE VALVE OPERATOR. THE COIL WAS REPLACED AND THE SYSTEM RETURNED TO SERVICE.
HPCS-MO-12	07/02/91	DURING PERFORMANCE OF TESTING, THE HIGH PRESSURE CORE SPRAY PUMP MINIMUM FLOW VALVE WOULD NOT CLOSE FROM THE CONTROL ROOM CONTROL SWITCH. THIS HAD NO EFFECT ON THE PLANT AS THE UNIT WAS IN A REFUELING OUTAGE AND THE SYSTEM WAS IN TEST. THE MOTOR PINION WAS SHEARED. THE MOTOR OUTPUT SHAFT WAS SHEARED IN HALF AND THE MOTOR STATOR WAS DAMAGED. CAUSE OF THIS DAMAGE IS UNKNOWN BUT IT APPEARS THAT SOMETHING HIT HPCS-MO-12. THE MOTOR AND MOTOR PINION WERE REPLACED. THE DECLUTCH MECHANISM WAS ALSO REPAIRED AND REPORTED SEPARATELY. ALL TESTING SATISFACTORY. SYSTEM RETURNED TO SERVICE .
HPCS-MO-12	07/05/91	DURING PERFORMANCE OF TESTING ON THE HIGH PRESSURE CORE SPRAY PUMP MINIMUM FLOW VALVE (HPCS-V-12) AFTER MOTOR REPLACEMENT, THE VALVE WOULD NOT OPERATE MANUALLY WITHOUT THE DECLUTCH MECHANISM BEING HELD DOWN.

Component ID	Failure Date	Description
		THIS HAD NO EFFECT ON THE PLANT AS THE UNIT WAS IN A REFUELING OUTAGE AND THE SYSTEM WAS IN TEST. THE TRIPPER SPRING WAS FOUND IN THE BOTTOM OF THE GEAR CASE AND THE TRIPPER FINGER CAPSCREWS WERE LOOSE. THE SPRING HOOK WAS BENT ON ONE END AND TWISTED. SUSPECT DAMAGE WAS CAUSED BY THE SAME OBJECT THAT DAMAGED THE MOTOR/MOTOR PINION (REPORTED SEPARATELY.) THE TRIPPER SPRING WAS REPLACED AND THE DECLUTCH MECHANISM WAS ADJUSTED. ALL TESTING SATISFACTORY. RETURNED SYSTEM TO SERVICE.
LD-MON-1A	01/08/91	WITH THE PLANT IN NORMAL FULL POWER OPERATION, ALARMS INDICATING HIGH REACTOR CORE ISOLATION COOLING (RCIC) TURBINE AND EQUIPMENT AREA HIGH TEMPERATURES WERE RECEIVED IN THE CONTROL ROOM. CONCURRENTLY, THE OUTBOARD ISOLATION VALVE FOR THE RCIC TURBINE STEAM SUPPLY, RCIC-V-8, AUTOMATICALLY CLOSED. REDUNDANT INSTRUMENTATION AND VISUAL OBSERVATION SHOWED NO ABNORMAL TEMPERATURES IN THE AREAS INDICATED BY THE ALARM. THIS RESULTED IN A LOSS OF THIS PORTION OF THE "A" TRAIN OF THE LEAK DETECTION INPUT TO THE NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM AND ALSO AN ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM. THERE WAS NO SIGNIFICANT PLANT EFFECT. THE CAUSE OF THE ISOLATION WAS TRACED TO A DEFECTIVE PRINTED CIRCUIT CARD IN THE LEAK DETECTION MONITOR WHICH PROVIDES INPUT TO THE RCIC COOLING LOGIC. THE CAUSE OF THE DEFECTIVE CARD IS UNKNOWN AT THIS TIME. THE DEFECTIVE CARD WAS REPLACED WITH SAME TYPE NEW AND THE SYSTEM WAS RETURNED TO SERVICE. VENDOR ANALYSIS OF THE CIRCUIT CARD HAS BEEN REQUESTED.
LD-MON-1A	09/18/91	RECEIVED ALARMS IN THE MAIN CONTROL ROOM INDICATING

Component ID	Failure Date	Description
		REACTOR WATER CLEANUP (RWCU) SYSTEM HEAT EXCHANGER ROOM HIGH TEMPERATURE, LEAK DETECTION (LD) SYSTEM HIGH DIFFERENTIAL FLOW RATE AND LD SYSTEM DIVISION 1 TROUBLE. RWCU OUTBOARD ISOLATION VALVE RWCU-V-4 AUTO CLOSED AND RWCU-P-1B TRIPPED. THE CLOSURE OF RWCU-V-4 IS AN ENGINEERING SAFETY FEATURE ACTUATION. NO PLANT EFFECT, LOSS OF SYSTEM A WALKDOWN AND INVESTIGATION WERE PERFORMED ON THE RWCU SYSTEM TO VERIFY SYSTEM INTEGRITY, NO LEAKAGE WAS FOUND. UPON FURTHER INVESTIGATION BY I&C TECHNICIANS IT WAS DISCOVERED THAT LD-TE-3E DISPLAY WAS READING UPSCALE. THIS IS THE RWCU HX ROOM SENSOR. THE UPSCALE READING OF LD-TE-3E WAS CAUSED BY A FAILED THERMOCOUPLE INPUT MODULE IN LD-MON-1A. CAUSE OF THE MODULE FAILURE IS UNKNOWN. IMMEDIATE CORRECTIVE ACTION WAS TO RESTORE THE RWCU SYSTEM TO SERVICE AFTER A THOROUGH INVESTIGATION. THE FAILED INPUT MODULE "A3" CARD (PIECE PART OF LD-MON-1A) IN LD-MON-1A WAS REPLACED. DISCUSSIONS WITH GENERAL ELECTRIC ARE ONGOING AS THIS IS THE SECOND SUCH FAILURE AT WNP-2 AND THE VENDOR'S ANALYSIS OF THIS NEW FAILURE WILL BE EVALUATED FOR NECESSARY ADDITIONAL CORRECTIVE ACTIONS .
MS-AO-3B	09/30/91	DURING PERFORMANCE OF TESTING IN PREPARATION FOR PLANT STARTUP FOLLOWING THE ANNUAL REFUELING OUTAGE, MAIN STEAM RELIEF VALVE 3B WOULD NOT OPEN FROM THE CONTROL SWITCH . THIS HAD NO PLANT OR SYSTEM EFFECT SINCE TECHNICAL SPECIFICATIONS REQUIRE THAT ONLY 12 OF THE 18 VALVES MUST BE OPERABLE. THE FAILURE WAS THE RESULT OF PHYSICAL DAMAGE TO THE AMPHENOL CONNECTOR (WIRES PULLED OUT) WHICH IS A PIECE PART OF THE AIR OPERATOR. THIS DAMAGE APPARENTLY OCCURRED FROM SOMEONE STEPPING

Component ID	Failure Date	Description
		ON THE CONNECTOR DURING A ROUTINE DRYWELL INSPECTION FOR LEAKS WITH THE REACTOR AT RATED PRESSURE. THE AMPHENOL CONNECTOR WAS REPAIRED AND TESTED.
MS-DPIS-11D	09/24/91	DURING PERFORMANCE OF SURVEILLANCE PROCEDURE TO CALIBRATE MAIN STEAM LINE 'D' HIGH DIFFERENTIAL PRESSURE SWITCH MS-DPIS-11D, NUMEROUS SPURIOUS HALF SCRAMS WERE GENERATED. THIS HAD NO SIGNIFICANT PLANT EFFECT AS THE UNIT WAS SHUTDOWN FOR REFUEL. THIS DID, HOWEVER, ISOLATE THE 'D' MAIN STEAM LINE. THE CAUSE OF THE FAILURE WAS NOISE GENERATED NEAR OR AT THE TRIP SETPOINT OF THE SWITCH. THIS NOISE WAS GENERATED BY THE HIGH LEVEL SWITCH. (PIECE PART OF MS-DPIS-11D). THE NOISE GENERATED ALSO CAUSED SPURIOUS SPIKES AND TRIPS ON IRM 'D', SRM 'D' AND IRM 'H'. REPLACED THE HIGH LEVEL MICROSWITCH. PERFORMED SCOPE TESTS AND CHANNEL CALIBRATION. ALL TEST RESULTS WERE SATISFACTORY.
MS-DT-1B	10/01/91	THE LOW PRESSURE TURBINE EXHIBITED HIGH VIBRATION ON THE NUMBER 6, 7, AND 8 BEARINGS AS INDICATED ON THE PLANT COMPUTER DURING UNIT SHUTDOWN FOR AN UNRELATED COMPONENT FAILURE. THE CAUSE OF THE FAILURE WAS AN IMBALANCE AT THE GENERATOR END OF THE TURBINE. WEIGHTS WERE INSTALLED ON THE ROTOR HUB UNDER THE DIRECTION OF A TURBINE VENDOR REPRESENTATIVE.
MS-DT-1C	10/01/91	THE LOW PRESSURE TURBINE EXHIBITED HIGH VIBRATION ON THE NUMBER 6, 7, AND 8 BEARINGS AS INDICATED ON THE PLANT COMPUTER DURING UNIT SHUTDOWN FOR AN UNRELATED COMPONENT FAILURE. THE CAUSE OF THE FAILURE WAS AN IMBALANCE AT THE GENERATOR END OF THE TURBINE. WEIGHTS WERE INSTALLED ON THE ROTOR HUB UNDER THE

Component ID	Failure Date	Description
		DIRECTION OF A TURBINE VENDOR REPRESENTATIVE.
RCIC-MO-45	07/29/91	DURING PERFORMANCE OF THE TWO YEAR POSITION INDICATION VERIFICATION SURVEILLANCE PROCEDURE ON THE REACTOR CORE ISOLATION COOLING TURBINE INLET MOTOR OPERATED VALVE , THE VALVE FAILED TO CLOSE FULLY AND ONLY OPENED TO 80 PCT. DIAGNOSTIC TESTING REVEALED THAT THE OPEN LIMIT SWITCHES WERE OUT OF ADJUSTMENT AND PHYSICAL INSPECTION REVEALED DAMAGED LIMIT SWITCH WIRING/LUGS ON POINTS 4, 9C, 14, AND 15 OF THE LIMIT SWITCH. (THE DAMAGED WIRING IN ITSELF DID NOT PREVENT THE VALVE FROM PERFORMING AS REQUIRED.) THE DAMAGED LIMIT SWITCH WIRING (COSMETIC) WAS REPAIRED AND THE OPEN LIMIT SWITCH WAS ADJUSTED SO THAT THE VALVE OPENED 90 TO 94 PERCENT. ALL TESTING RESULTS WERE SATISFACTORY AND THE SYSTEM WAS RETURNED TO SERVICE.
RCIC-MO-45	09/27/91	DURING PERFORMANCE OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM OPERABILITY TEST, THE 4-INCH MOTOR OPERATED TURBINE INLET VALVE (RCIC-MO-45) WOULD NOT OPEN ON DEMAND. THIS HAD NO EFFECT ON THE PLANT BUT RESULTED IN THE LOSS OF THE RCIC SYSTEM. THE CAUSE OF THE FAILURE WAS MIS-WIRING OF THE LIMIT SWITCH OF THE 10-INCH MOTOR OPERATED TURBINE EXHAUST TO THE SUPPRESSION POOL (RCIC-V-68) VALVE WHICH PROVIDES AN OPEN PERMISSIVE TO RCIC-MO-45. THIS MIS-WIRING WAS DONE DURING RETERMINATION OF THE LIMIT SWITCH WIRING FOLLOWING VALVE REFURBISHMENT. THE LIMIT SWITCH MIS-WIRING OF RCIC-MO-68 WAS CORRECTED AND THE OPERABILITY TEST WAS COMPLETED SATISFACTORILY.
RFW-DPT-4A	06/10/91	REACTOR FEEDWATER LEVEL INDICATOR (RFW-LI-606A), WHICH

Component ID	Failure Date	Description
		RECEIVES IT'S SIGNAL FROM RFW-DPT-4A WAS NOTED TO DRIFT DOWN TO 42" WHILE CURRENT PLANT CONDITIONS WOULD HAVE OVERRANGED THE INDICATOR. THIS WAS OBSERVED A NUMBER OF TIMES BY OPS PERSONNEL WHILE COMPLETING SHIFT LOGS. NO PLANT EFFECT AS THE UNIT WAS SHUTDOWN FOR REFUEL. THIS CHANNEL OF THE ISOLATION INSTRUMENTATION WAS DECLARED INOPERABLE. THE CAUSE OF THE FAILURE WAS A DAMAGED FIELD CABLE PROVIDING INPUT TO THE LEVEL INDICATOR. CAUSE OF THE DAMAGE IS UNKNOWN. DETERMINATED AND SPARED THE BAD PAIR (#3) OF CABLE 01303/C34A-002 AT BOTH ENDS AND RETERMINATED WITH GOOD PAIR (#5) OF SAME CABLE. ALL SUBSEQUENT TESTING/INDICATIONS WERE SATISFACTORY.
RFW-LS-624B	09/29/91	DURING PERFORMANCE OF CHANNEL FUNCTIONAL TESTING BY INSTRUMENTATION AND CONTROL TECHNICIANS, REACTOR VESSEL HIGH LEVEL RFW/MAIN TURBINE TRIP SWITCH (RFW-LS-624B) WAS FOUND TO BE OUT OF CALIBRATION HIGH. THE CAUSE OF THE OUT OF CALIBRATION CONDITION IS UNKNOWN. THE SWITCH WAS CALIBRATED WITHIN SPECIFICATIONS AND ALL TESTING WAS SATISFACTORY.
RFW-MO-65A	10/03/91	WITH THE PLANT IN MODE 2 (STARTUP), THE REACTOR FEEDWATER 24" MOTOR OPERATED INLET TO THE REACTOR PRESSURE VESSEL WOULD NOT OPEN ON DEMAND. THIS MADE THE "A" TRAIN OF THE REACTOR FEEDWATER SYSTEM INOPERABLE AND PREVENTED CONTINUED PLANT STARTUP. THE CAUSE OF THE FAILURE WAS AN ACCUMULATION OF A FILM OF OIL ON THE CONTACTS OF THE LIMIT SWITCH WHICH PREVENTED THE CONTACTS FROM MAKING UP. THIS IS DUE TO THE ORIENTATION OF THE MOTOR OPERATOR WHICH REQUIRES THAT THE LIMIT SWITCH HOUSING FACE DOWN ALLOWING

Component ID	Failure Date	Description
		ACTUATOR GREASE TO MIGRATE OUT OF THE GEAR CASE ONTO THE LIMIT SWITCH ASSEMBLY. THE CONTACTS WERE CLEANED AND BURNISHED. THE OPERATOR THEN PERFORMED AS REQUIRED.
RFW-V-32B	04/26/91	DURING PERFORMANCE OF LOCAL LEAK RATE TEST ON THE REACTOR FEED WATER 'B' OUTBOARD ISOLATION VALVE, THE VALVE FAILED TO SEAT AND ISOLATE FLOW. THIS HAD NO EFFECT ON THE PLANT AS THE PLANT WAS SHUTDOWN FOR REFUELING AND THE SYSTEM WAS IN TEST. THE CAUSE OF THE FAILURE WAS WEAR OF THE RESILIENT SOFT SEAT. REPLACED THE RESILIENT SOFT SEAT, PRESSURE SEAL RING, AND FLEXIBLE GASKETS. REPACKED VALVE AND TORQUED BONNET. SUBSEQUENT PERFORMANCE OF LLRT WAS SATISFACTORY.
RHR-MO-68B	06/02/91	DURING ATTEMPTS TO TRANSFER SHUTDOWN COOLING LOOPS WITH THE PLANT IN COLD SHUTDOWN MODE, OPERATORS NOTED THE STANDBY SERVICE WATER ISOLATION VALVE TO THE RHR HEAT EXCHANGER DID NOT FULLY CLOSE UNDER SYSTEM DIFFERENTIAL PRESSURE. THIS RESULTED IN A DEGRADED LOOP IN THAT THE ABILITY TO ISOLATE WAS LOST, BUT THE LOOP COULD BE OPERATED. THERE WAS NO SIGNIFICANT PLANT EFFECT: TORQUE SWITCH SETTINGS WERE TOO LOW TO FULLY CLOSE VALVE AGAINST SYSTEM PRESSURE DUE TO UNKNOWN REASONS . TORQUE SWITCH SETTINGS WERE INCREASED AND THE OPERATOR WAS DYNAMICALLY TESTED.
RHR-V-112A	11/04/91	DURING A ROUTINE STARTUP 1000 PSIG INSPECTION OF THE REACTOR DRYWELL IN PREPARATION FOR PLANT RESTART , PLANT PERSONNEL DISCOVERED A SMALL LEAK IN LOOP A OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHR). ESTIMATED VOLUME WAS 20 DROPS PER MINUTE. THE LEAK WAS AT A WELDED CONNECTION OF A 3/4 INCH TEST

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (cont.)

Component ID	Failure Date	Description
		VALVE RHR-V-161A FOR THE SHUTDOWN COOLING RETURN VALVE RHR-V-112A. THIS CONDITION REPRESENTED PRESSURE BOUNDARY LEAKAGE AND RESULTED IN A PLANT SHUTDOWN AND DECLARATION OF AN UNUSUAL EVENT. THE CAUSE OF THE EVENT WAS A CONSTRUCTION FABRICATION DEFICIENCY (LACK OF INTERNAL FUSION ON THE INSIDE OF THE WELD) IN THE FILLET WELD AT THE PROCESS PIPE SOCKET FORGING WHICH MAKES UP THE NIPPLE CONNECTION TO DRAIN VALVE RHR-V-161A. THE FAILED CONNECTION WAS REMOVED AND REPLACED WITH A SHORT-COUPLE NIPPLE AND PIPE CAP. THE DRAIN CONNECTION WILL BE REDESIGNED AND REPLACED DURING THE 1992 MAINTENANCE AND REFUELING OUTAGE .
RPS-RLY-K16C	08/19/91	OPERATIONS PERSONNEL OBSERVED THAT THE REACTOR PROTECTION SYSTEM SHUTDOWN / SCRAM RESET RELAY (RPS-RLY-K16C) LOCATED IN THE MAIN CONTROL ROOM PANEL P609 WAS BUZZING LOUDLY WITH SMALL MOUNDS OF COPPER FILINGS ON THE CONTACTS. THIS HAD NO EFFECT ON THE PLANT AS THE UNIT WAS SHUTDOWN FOR REFUEL AND HAD NO EFFECT ON THE SYSTEM AS THE RELAY WAS REPLACED PRIOR TO FAILURE. CAUSE OF FAILURE IS NORMAL WEAR DUE TO AGING. CALIBRATED, TESTED, AND INSTALLED NEW TIME DELAY RELAY.
RPS-RLY-K9B	07/17/91	THE REACTOR PROTECTION SYSTEM TURBINE TRIP BYPASS RELAY K9B APPEARED TO BE OVERHEATING . THIS WAS OBSERVED BY PLANT PERSONNEL PERFORMING PREVENTIVE MAINTENANCE TASKS ON A RELAY ADJACENT TO THE K9B RELAY. THIS HAD NO PLANT EFFECTS SINCE THE PLANT WAS SHUTDOWN. THE CAUSE OF THE RELAY FAILURE / OVERHEATING IS NOT KNOWN AT THIS TIME. THIS RELAY IS A NORMALLY ENERGIZED RELAY AND THE FAILURE MAY BE A RESULT OF HEAT BUILDUP

2.4 SIGNIFICANT CORRECTIVE MAINTENANCE PERFORMED ON SAFETY-RELATED EQUIPMENT (cont.)

Component ID	Failure Date	Description
		OVER TIME. REPLACED THE RELAY WITH A NEW AUXILIARY RELAY .
SW-MO-12A	08/16/91	THE STANDBY SERVICE WATER PUMP DISCHARGE MOTOR OPERATED VALVE (SW-MO-12A) BLEW MAIN FUSES WHILE ATTEMPTING TO GO CLOSED ON DEMAND FROM MAIN CONTROL ROOM. THE PLANT WAS SHUTDOWN FOR REFUELING AT THE TIME AND THE SYSTEM WAS IN TEST TO ALLOW FOR VALVE OPERATOR DIAGNOSTIC TESTING ON SW-MO-12A. DISASSEMBLY OF THE MOTOR OPERATOR REVEALED THAT THE TORQUE SWITCH HAD BEEN DAMAGED, THE SHAFT SLEEVE WAS BROKEN, LOWER ROLLER BEARING CUP WAS DAMAGED , HOUSING NUTS HAD BEEN STRETCHED AND THE WORM GEAR WAS BENT. THE CAUSE OF THE DAMAGE WAS DUE TO OVERTORQUEING OF OPERATOR DURING TESTING. REPLACED ALL DAMAGED COMPONENTS. PERFORMED TESTING, ALL RESULTS SATISFACTORY. COMPONENT RETURNED TO SERVICE.
SW-MO-24A	05/05/91	DURING PERFORMANCE OF TWO YEAR VALVE POSITION INDICATION VERIFICATION SW-MO-24A, MOTOR OPERATED SERVICE WATER INLET TO THE 'A' RESIDUAL HEAT REMOVAL HEAT EXCHANGER, FAILED ITS STROKE TIME TECHNICAL SPECIFICATION REQUIREMENT TEST DURING OPERATOR DIAGNOSTIC TESTING. IT WAS DISCOVERED THAT THE OPEN LIMIT SWITCH ON SW-MO-24A WAS OUT OF ADJUSTMENT. ADJUSTED OPEN LIMIT SWITCH TO PROPER SETTING (90% TO 96% OF VALVE FULL OPEN POSITION). PERFORMED PPM 7.4.0.5.16 AND VERIFIED PROPER VALVE POSITION INDICATION AND STROKE TIMES .
SW-V-165B	02/19/91	AN OPERATOR FOUND DURING ROUNDS THAT THE 18" BUTTERFLY BYPASS VALVE FOR SERVICE WATER SPRAY POND "B" SPRAY

Component ID	Failure Date	Description
		RING LEAKED BY AND WAS VERY HARD TO OPERATE . THIS COULD HAVE AFFECTED TEMPERATURE CONTROL IN THE IN SERVICE STANDBY SERVICE WATER TRAIN . UPON DISASSEMBLY OF THE VALVE THE RUBBER SEAT WAS FOUND DAMAGED (1 1 / 2") MISSING AT APPROX. 12:00 POSITION), TAPER PIN THROUGH STUB SHAFT HAD BEEN SHEARED, SEAL WASHERS WERE MISSING FROM UNDER TAPER PIN JAM NUTS, AND SEAL RING SEATING SURFACE HAS A WORN (FLAT) SPOT APPROX. 1 1/2" LONG AS WELL AS VARIOUS PITTED AND SCRATCHED AREAS ON SEATING SURFACES. DISASSEMBLED VALVE, CLEANED SEAL RING AND DISC SEATING SURFACES, INSTALLED NEW RUBBER SEAT ASSEMBLY, FABRICATED AND INSTALLED NEW SPRING PIN USING TYPE 416SS UPGRADED MATERIAL, REPLACED BROKEN TAPER PIN, INSTALLED NEW THREADSEAL WASHERS, AND TIGHTENED TAPER PIN SNUG TIGHT. ALSO DISASSEMBLED GEAR OPERATOR, CLEANED, REPLACED GREASE AND REASSEMBLED .

2.5 FUEL PERFORMANCE

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

INTRODUCTION

Fuel cycle off-gas data indicated no leaking fuel during Cycle 6 operation. The off-gas rate remained approximately constant at 7000 μCi/sec throughout the cycle. This off-gas was attributed to fuel leakage experienced in the previous cycle and was determined to be recoil only. All leaking fuel elements were discharged at the end of the previous cycle but a strong residual component remains in the core. This component has been rather conclusively identified with the major leaking fuel rod from the previous cycle by correcting the sum-of-six data for the known plutonium content of that fuel rod.

Inspection of a representative sample of discharged fuel elements and one fuel channel was carried out by Supply System personnel. GE personnel assisted in this inspection and provided the inspection setup. The inspection consisted of a visual examination of the two sides of the selected discharged fuel elements and channel.

INSPECTION RESULTS

A total of eight fuel elements and one channel discharged at the end of Cycle 6 were inspected. No evidence of mechanical damage, geometric distortion, or rod bow was observed. All fuel rods inspected appeared properly seated in the lower tie plate. All spacers were in proper position. The spacer side plate nodular corrosion buildup ranged from virtually none to perhaps 50% coverage with small nodules making these side plates the cleanest so far for WNP-2 discharged fuel. One instance of seal spring distortion, minor in extent, was observed. The type of distortion observed suggested that the damage could well have occurred during fuel assembly dechanneling. The fuel rod surface was generally covered with a heavy uniform crud, preventing detailed observation of the fuel clad surface. In isolated instances, dechanneling scraped the crud from the clad surface revealing a clean surface for inspection. In these instances little or no nodular activity was observed on the clad surface.

The fuel channel inspected, the highest exposed ASEA Atom channel discharged from WNP-2 to date, exhibited a uniform coating of light colored oxide material on unwelded surfaces. The heat effected zone of the weld surface was clean. There was no evidence of mechanical damage to the channel.

2.6 10CFR50.59 CHANGES, TESTS, AND EXPERIMENTS

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

2.6.1 PLANT MODIFICATIONS

Permanent Plant Modifications at WNP-2 are implemented with a Plant Modification Request (PMR). The following PMRs implemented in 1991 required a Safety Evaluation in accordance with 10CFR50.59. Each permanent change was evaluated and determined not to represent an Unreviewed Safety Question nor require a change to the WNP-2 Technical Specifications.

2.6.1.1
PMR 85-0743

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

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

Safety Evaluation Summary

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

function of the valve did not change and probability of failure decreased.

2.6.1.2
PMR 86-0283

This modification installed a liquid level indicator on expansion tank piping at CCH-CR-1B. To accomplish this, CCH-CR-1B had to be out of service for approximately two days at the same time CCH-CR-1A was out of service for repair. This resulted in both trains of chillers being out of service at the same time.

Safety Evaluation Summary

The emergency control room chillers, CCH-CR-1A and 1B, are mitigating systems for post accident operation. Therefore, the probability of the occurrence or consequences of a previously evaluated LBD accident were not increased. These chillers provide additional cooling capacity for personnel comfort. The Standby Service Water System (SW), on its own, provides adequate control room cooling for equipment operability. This activity was performed with the Plant at Cold Shutdown in accordance with the Technical Specification requirements. Past experience has shown that control room temperature does not exceed 85° F even with the chillers not operating. Thus, the loss of the chillers did not increase the likelihood of the malfunction of equipment important to safety or its consequences. Finally, as Standby Service Water ensured that the 104 degree (F) limit for control room equipment was not exceeded, the margin of safety for the equipment was not reduced.

2.6.1.3
PMR 86-0323

PMR 86-0323 was issued to perform modifications to RHR-V-3A, 3B and RHR-V-48A, 48B to better balance the flow rate through the RHR heat exchanger and to increase reliability, respectively.

These Residual Heat Removal valves (RHR-V-3A & 3B) are gate valves. They are safety related and are required to establish/control the flow path through the RHR heat exchangers which in turn control heat removal from the reactor vessel. The gate valves currently installed as RHR-V-3A & 3B are not considered the optimum design for throttling service. During shutdown cooling mode of RHR operation, the cool-down rate is controlled by varying the percentage of water that goes through the heat exchanger versus the amount of water that bypasses the heat exchanger. PMR 86-0323 changed RHR-V-3A & 3B to globe valves which are better suited for throttling service.

RHR-V-48A & 48B were modified due to a recommendation from their

function of the valve did not change and probability of failure decreased.

2.6.1.2
PMR 86-0283

This modification installed a liquid level indicator on expansion tank piping at CCH-CR-1B. To accomplish this, CCH-CR-1B had to be out of service for approximately two days at the same time CCH-CR-1A was out of service for repair. This resulted in both trains of chillers being out of service at the same time.

Safety Evaluation Summary

The emergency control room chillers, CCH-CR-1A and 1B, are mitigating systems for post accident operation. Therefore, the probability of the occurrence or consequences of a previously evaluated LBD accident were not increased. These chillers provide additional cooling capacity for personnel comfort. The Standby Service Water System (SW), on its own, provides adequate control room cooling for equipment operability. This activity was performed with the Plant at Cold Shutdown in accordance with the Technical Specification requirements. Past experience has shown that control room temperature does not exceed 85° F even with the chillers not operating. Thus, the loss of the chillers did not increase the likelihood of the malfunction of equipment important to safety or its consequences. Finally, as Standby Service Water ensured that the 104 degree (F) limit for control room equipment was not exceeded, the margin of safety for the equipment was not reduced.

2.6.1.3
PMR 86-0323

PMR 86-0323 was issued to perform modifications to RHR-V-3A, 3B and RHR-V-48A, 48B to better balance the flow rate through the RHR heat exchanger and to increase reliability, respectively.

These Residual Heat Removal valves (RHR-V-3A & 3B) are gate valves. They are safety related and are required to establish/control the flow path through the RHR heat exchangers which in turn control heat removal from the reactor vessel. The gate valves currently installed as RHR-V-3A & 3B are not considered the optimum design for throttling service. During shutdown cooling mode of RHR operation, the cool-down rate is controlled by varying the percentage of water that goes through the heat exchanger versus the amount of water that bypasses the heat exchanger. PMR 86-0323 changed RHR-V-3A & 3B to globe valves which are better suited for throttling service.

RHR-V-48A & 48B were modified due to a recommendation from their

manufacturer to increase the weld size of the connection between the disc and disc skirt. This was accomplished by having the weld connecting the disc to the disc skirt increased to a 3/16" fillet for 360 degrees around the joint. This increased weld size will make these valves more reliable.

Safety Evaluation Summary

The Safety Evaluation concluded the proposed changes did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety because (1) the modification to the welds on RHR-V-48A & 48B will serve to increase their reliability and (2) the safety function of RHR-V-3A & 3B is unchanged by replacing the existing gate valves with globe valves and will provide more reliable service.

2.6.1.4
PMR 86-0352

This PMR was initiated to reduce repair of the DC soakback pumps. Maintenance requirements were eased without reducing the reliability of the Diesel Generator or introducing a new event which had not been analyzed.

To accomplish this, new AC soakback pumps (DLO-P-11A1, 11A2, 11B1, 11B2 & 12) were added to the Diesel Lube Oil (DLO) system. The existing DC motor driven soakback pumps (DLO-P-2A1, 2A2, 2B1, 2B2 & 10) are now used as backup in the event the AC pumps are unavailable. Control logic was added so the DC pumps start when the AC pump discharge pressure at the soakback filter or on the discharge of the HPCS AC soakback pump falls below 7 psi. Logic is also included to shut off the DC pump if the AC pump starts again so that high lube oil pressure in the soakback system does not cause oil to enter the engine circulating lube oil system through the 75 psig check valve. Excess oil may lead to foaming in the diesel main bearings causing premature failure or result in "hydraulic locking" of a piston do to oil accumulation.

Safety Evaluation Summary

The proposed modification does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety because the soak back pumps perform the non-safety related function of circulating lube oil through the turbocharger during shutdown to minimize wear on startup and shutdown. The added AC pumps provides additional assurance that unscheduled repair/replacement of the DC motors will be avoided. The extra pump also provides added assurance that the turbocharger will not experience excess wear.

2.6.1.5
PMR 86-0621

PMR 86-0621 was issued to delete equipment associated with the Floor/Equipment Drain Leak Detection System by removing the present restricting orifices, level switches and deactivating Board E in the control room. This modification also reworked the indication circuit so both indication and annunciation functions would be more serviceable.

The purpose of annunciator drop S1-2.4 was to aid in the determination of the location of a leak. The window was alarmed by a series of contacts throughout the Class II equipment and floor drain systems. The drains contained a restricting orifice which would limit flow and cause the fluid to back-up and trip limit switch, setting off the alarm. The fluid would bypass to a second line when it backed up far enough, the switch could reset, and the cycle started over again if the flow was low enough. This was problem was aggravated by small debris clogging the orifices. The alarm was constantly coming in to the Control Room due to minor leaks that could not be repaired at power. The system was virtually useless as configured.

Safety Evaluation Summary

The systems affected by these modifications are not safety related. Consequences of any spill or leak will not change because of the implementation of PMR 86-0621. Mitigation of small spills or leaks requires a plant walkdown to specifically identify problems. Sump pump run times still provide an indication of where heavy leakage is occurring. The Safety Analysis found that the proposed changes did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety.

2.6.1.6
PMR 87-0151

The purpose of manual valves EDR-V-18 & FDR-V-15 is to provide an isolation boundary to support testing of the downstream containment isolation valves. Containment isolation is the purpose of air operated valves FDR-V-3 & FDR-V-4. These four gate valves leaked excessively during leak rate testing.

This Plant Modification replaced the existing, installed gate valves with ball valves to eliminate the excessive leakage. The new valves should also require less maintenance. In addition, several unnecessary snubbers near these valves were eliminated.

Safety Evaluation Summary

Based upon the safety analysis, neither plant functions nor the purpose of the valves (manual and air operated) is being changed as a result of this design change. The probability of occurrence or consequences of an accident, or malfunction of equipment important to safety is not increased by these changes. The modification will

not create an accident or malfunction of a different type than those evaluated previously in the Licensing Basis Documents (LBDs). The margin of safety was not reduced as defined in the bases for pertinent Technical Specifications.

2.6.1.7
PMR 87-0282

Fan coil units RRA-FC-8 and 9 are located within the Main Steam Tunnel. The two units normally operate continuously in recirculation mode to remove heat generated by the steam piping in the tunnel.

It was determined that a third backup fan coil unit be installed to be available in case either RRA-FC-8 or RRA-FC-9 become unavailable. PMR 87-0282 detailed the addition of RRA-FC-21 as that backup unit.

Safety Evaluation Summary

The Main Steam Tunnel fan coil unit Heating Ventilation Air Conditioning (HVAC) system is not safety related. The safety analysis stated that the proposed change did not increase the probability of occurrence or consequence of an accident, or malfunction of equipment important to safety. In fact, the additional fan cooler provides added assurance that the normal steam tunnel temperatures are maintained within the allowable limits.

2.6.1.8
PMR 88-0038

Redundant equipment existed within the Drywell (D/W), Wetwell (W/W) Temperature Monitoring Systems and the Primary Containment Pressure System. D/W and W/W temperature data was transmitted to additional panels prior to being paralleled to provide data at Control Room panel P601. D/W pressure data was displayed on two recorders per division at panel P601.

Via PMR/BDC 88-0038, existing redundant Regulatory Guide 1.97 Category 2 Drywell and Wetwell temperature monitoring instrumentation was deleted at Panels P831 and P814, respectively. All drywell and wetwell temperature information was consolidated at Panel P601. Redundancy is not required for Category 2 variables. All three drywell pressure signals were combined to a single recorder.

Additional modifications made under PMR/BDC 88-0038 were: new data transmission cable was installed which connects the scanners and recorders, two new Regulatory Guide 1.97 temperature indicators were added in response to human factors concerns for post-accident D/W and W/W Temperature Averages, a downgrade of D/W and W/W

temperature instrumentation Quality Class (to QC 2+) was performed utilizing EI 2.31. The Final Safety Analysis Report (Appendix F Table 4.1 and Figure 76) was revised as required to show new equipment and cabling which provides W/W temperature data for plant shutdown.

Safety Evaluation Summary

The changes made under PMR 88-0038 did not create the possibility of an accident or malfunction of equipment important to safety or a different type than previously evaluated in the Licensing Basis Documents. Technical Specification bases were neither compromised nor reduced. The new data transmission cable , while not IEEE-383 fire tested cable, was tested and found to conform to fire propagation and smoke optical density criteria as listed in the National Electrical Code. Finally, no system information was deleted and no equipment/signal reliability/accuracy was reduced.

2.6.1.9
PMR 88-0338

Existing air operated dampers WOA-AD-1A and 1B are part of the Radwaste Building HVAC system. PMR 88-0338 replaced them with backdraft dampers at the discharge of fans WOA-FN-1A and 1B.

Safety Evaluation Summary

This design change did not increase the consequences of an accident or equipment malfunction which had been previously evaluated in the licensing basis documents. The Radwaste HVAC system is not safety related. The replacement of the existing air operated dampers with backdraft dampers will increase the system's reliability. The previously existing air operated dampers were not addressed in the Technical Specifications.

2.6.1.10
PMR 88-0353

Nonconformance Report (NCR) 289-0019 documented several issues with regard to the Containment Nitrogen (CN) system whereby equipment essential to safety could potentially be damaged by the temperatures or oxygen-deficient environment associated with liquid or gaseous nitrogen following a CN system line break.

The CN system is a Quality Class 2, non-safety related system. This modification created a Quality Class 1, safety related "island" in both the low flow side and high flow side of the nitrogen skid. This provided a reliable, single failure proof isolation of liquid or very cold nitrogen regardless of downstream conditions (ie., a pipe break or loss of pressure or flow control functions). Piping connected to the tank up to and including the new isolation valves was analyzed to assure that all pipe stresses

were below those allowed for Seismic Class I piping.

Safety Evaluation Summary

This modification did not increase the likelihood of any accident or equipment malfunction evaluated in the Licensing Basis Documents. It was specifically designed not to challenge design safety systems. The PMR was written to address a newly postulated unanalyzed condition. The change did not affect the Technical Specifications.

2.6.1.11
PMR 89-0034

PMR/BDC 89-0034 provided for the utility connections for the new site office building (Building 26). The connections are for fire protection, potable water, and sewer. The BDC also puts a cap on the industrial area service air system.

Safety Evaluation Summary

The safety evaluation concluded that the design change did not significantly increase the likelihood or the consequences of an accident or equipment malfunction which had been previously evaluated in the Licensing Basis Documents. The modification of the Fire Protection and Potable Water systems was done in accordance with design specifications and codes. The work was done in the site industrial area and did not affect any existing safety analysis. The connection to the Plant Fire Protection system was provided with isolation capability.

2.6.1.12
PMR 89-0086

This PMR added a drop chute in the Control Rod Drive (CRD) repair room in which canisters containing CRD filters pass through the floor at Elevation 471′ and into the Railroad Bay below for storage until job completion. Also added were a new 480 volt power panel to provide power for the CRD rebuild room and related equipment and a 10′ X 10′ structure at the entrance to the CRD rebuild room to assist in the control of airborne contamination.

Safety Evaluation Summary

The conclusion of the Design safety analysis was that the change did not increase the probability of a occurrence or consequences of an accident, or malfunction of equipment important to safety. The drop chute is Class I because it is a part of secondary containment when the railroad bay doors are open. A valve located in the CRD repair room is designed to always isolate secondary containment regardless of its position. Thus, this modification does not jeopardize secondary containment integrity.

2.6.1.13
PMR 89-0159

The purpose of the Class 1E 120 AC volt power supply system is to provide electrical power for all the safety related devices and systems which are supplied from the Class 1E, 120 volt Instrument and Control Power Distribution Panels. The existing design provides transformers to serve the 120 volt panels. Minimum voltage at the 120 volt level was analyzed and it could not be assured that all devices would be supplied with a voltage level within their rated voltage range if source voltage to the transformers were degraded to just above the setpoint of the degraded voltage transfer relays on the 4160 volt system.

This modification added voltage regulators in the power supply circuit for each of the Class 1E 120 volt power panels. The regulators maintain rated voltage at the panels.

Safety Evaluation Summary

This change will not increase the probability of occurrence or consequences of an accident, or malfunction of equipment important to safety as it will eliminate the possibility that the 120 volt safety related devices might fail to function because of insufficient voltage being supplied to the devices when the power source is degraded.

2.6.1.14
PMR 89-0289

The safety function of the 480V feeders to subfed Motor Control Centers (MCCs) is to passively carry safety related loads. The cable jumper (or Class 1E cable splice) previously installed extends the subfeeder circuit to the primary source MCC bus side connections. The cable jumpers (or Class 1E cable splices) do not provide any normal protective circuit functions.

PMR 89-0289 replaced the previously installed Class 1E splices (or cable jumpers) with new General Electric molded case circuit breakers which restored and upgraded normal power system circuit protective functions.

Safety Evaluation Summary

Previous evaluations in the LBDs cover single failure on one of the redundant Class 1E divisions of the AC Power Distribution System. Included were the consequences of losing one entire safety division function that would exceed (and therefore bound) any impact to plant safety from the postulated malfunctions of the components installed by PMR/BDC 89-0289.

2.6.1.15
PMR 89-0299

Main Steam Pressure Transmitters MS-PT-8A, B, & C were installed to monitor condenser pressure. This PMR relocated these pressure transmitters, rescaled the attendant pressure indicators and added piping supports inside the condenser.

Safety Evaluation Summary

This change was Class II, Seismic 2. It did not degrade or affect any Class I system. The Unanalyzed Safety Question analysis was required by procedure because the FSAR Figure 3.2-23A was changed to show the new pressure tap locations for the transmitters. The installation of the supports inside of the condenser will lessen the probability of an already analyzed condition (condenser tube failure).

2.6.1.16
PMR 90-0026

Containment Instrument Air (CIA) Compressors 1A and 1B were installed originally to be the backup system for the cryogenic nitrogen source. The CIA compressors were eliminated as a supply source by capping the common discharge line during the 1988 refueling outage. Safety related accumulators and nitrogen bottles now serve as the backup supply source.

PMR/BDC 90-0026 removed the CIA compressors, associated piping, hangers and concrete pads.

Safety Evaluation Summary

This change did not increase the possibility of occurrence or consequences of an accident, or malfunction of equipment important to safety because (1) the CIA compressors are not safety related and (2) physical removal of the CIA compressors will not increase the probability of an accident since the compressors were not being used or credited in the accident analyses.

2.6.1.17
PMR 90-0081

Cables 2DG2-26, 36, 44, 45, BSYNC-9028, 2SM8-131, 1SM7-121, 2M88A-145 and 2D12D-4 support normal operation and to provide Appendix R shutdown functions. They are not isolated from the effects of a Main Control Room fire. This PMR provided isolation for the above-mentioned cables.

Safety Evaluation Summary

The modification did not increase the possibility of occurrence or

consequences of an accident, or malfunction of equipment important to safety as the change reduces the possibility of accidents or equipment malfunctions by providing additional isolation between certain design basis fires and those systems necessary to safely shutdown the reactor.

2.6.1.18
PMR 91-0226

EES-5 was issued to revise the WNP-2 fuse and motor overload selection criteria. An evaluation of the fuses and overload heaters found some not in agreement with EES-5. PMR 91-0226 requested a BDC to provide information to change fuses and Thermal Overload Heaters (TOLs) for Class 1E motors as required by EES-5. In addition, TOL heaters for non-1E motors connected to Class 1E MCCs having less than zero per cent margin at 90% of normal voltage were changed to agree with EES-5.

The objective of this effort is to provide fuses sized to protect motor branch circuit conductors and TOL heater elements sized to assure operation of Class 1E motors during accidents and low bus voltage conditions.

Safety Evaluation Summary

The design safety analysis found that modification would not increase the probability of occurrence or consequences of a previously analyzed accident, or malfunction of equipment important to safety. In fact, it would reduce the probability that the TOL relay function could interfere with the safety function of the motor.

2.6.1.19
PMR 86-0395

PMR/BDC 86-0395 installed three new fire hydrants and relocated one other at the site warehouse complex. Additionally, a new fire main was extended around the warehouses.

Safety Evaluation

The addition of the new hydrants and fire main will not increase the probability of occurrence or consequences of an accident, or malfunction of equipment important to safety because the fire protection system is not safety related and has no direct interface with Quality Class I systems. These improvements will increase the reliability of the Fire Protection System. The changes were made in accordance with the NFPA Code and do not impact Plant Fire Protection capabilities.

2.6.1.20
PMR 87-0158

This PMR installed two upstream pressure regulating valves, a detector check valve assembly and associated piping in the Circulating Water Pumphouse. The modification provided a method of monitoring the leakage rate of the site Fire Protection system and a way of controlling its pressure. The Fire Protection system is normally maintained in a pressurized state by a small line from the Circulating Water system.

Safety Evaluation Summary

This plant modification did not involve an Unreviewed Safety Question or result in a change to the WNP-2 Technical Specifications because (1) the Fire Protection System is Quality Class 2+ and does not perform a safety function and (2) the addition of the valves and piping does not affect any margin of safety discussed in the Technical Specifications. The changes made increase the level of control and monitoring capability available.

2.6.1.21
PMR 87-0284

The existing control configuration started up the Control Room Emergency Chillers in conjunction with the Standby Service Water Pump. This ensured that the air handler to the Control Room was supplied with coolant during an emergency. However, the existing configuration unnecessarily started the chillers during non-emergency conditions. This tended to draw non-condensables into the Chiller thereby degrading its performance.

PMR 87-0284 installed an off-auto switch in the Control Room. This transferred the responsibility for starting the Chillers from automatic signals to the Control Room Operators. They will make the decision whether or not the Chillers are to be started based on Plant and Control Room conditions.

Safety Evaluation Summary

This change does not affect the operation of the Chillers. It only affects how and when the Emergency Chillers are to be started. By preventing the Chillers from operating during non-accident conditions, the wear on the Chillers will decrease and the likelihood of a malfunction is lessened.

2.6.1.22
PMR/BDC 88-0003

Z signal trip units are radiation monitors that have Geiger-Mueller (GM) detectors mounted on the Reactor Building ventilation exhaust plenum. These monitors provide trip signals to failsafe relay logic in relay cabinets 1 and 2 in response to high radiation in the plenum.

This modification moved the Z signal trip units' power supplies from their existing Reactor Protection System (RPS) bus feeds to divisional Uninterruptable Power Source (UPS) bus feeds. This reduces the potential for spurious safety system actuation due to loss of power. Discrepancies between the existing configuration and the WNP-2 design requirements for failsafe circuits were corrected by this modification. PMR 88-0003 also corrected electrical separation, equipment identification, and failsafe design discrepancies identified in Licensee Event Report (LER) 89-039 and Nonconformance Report (NCR) 289-0747 and justified the routing of a portion of the sensor-to-trip unit cable in a non-failsafe raceway.

Safety Evaluation Summary

This PMR did not increase the likelihood of any accident or equipment malfunction which had been previously analyzed in the Licensing Basis Documents. After the modification, a malfunction of a trip unit will result in actuation of that unit's contribution to a two-out-of-two trip signal (failsafe design). This removes a discrepancy between the existing equipment and WNP-2 design criteria for failsafe design as well as reducing the potential for spurious safety system actuation by increasing the reliability of the power sources. There are no system functional changes. The change did not create an accident or malfunction not analyzed previously. Finally, the change did not affect the Technical Specification safety margins or require a Technical Specification change.

2.6.1.23
PMR 88-0242

Plant Modification 88-0242 removed air operated dampers TEA-AD-15A and 15B, associated actuators and controls from the Turbine Building Exhaust Air (TEA) System. The dampers did not operate properly. The existing volume dampers were retained to control the air flow rate.

Safety Evaluation Summary

The safety evaluation determined that the removal of air operated dampers TEA-AD-15A and 15B and the use of existing volume dampers to control air flow rates did not change the function of the TEA system and would not increase the consequences of an accident or malfunction of equipment important to safety. This system is not safety related. TEA-AD-15A and 15B are not discussed in the Technical Specifications.

2.6.1.24
PMR 88-0254

Carbon steel structured shapes support stainless steel Division II sensing lines. These lines are pressure sensing lines coming from

the Residual Heat Removal (RHR), Containment Monitoring (CM) and Main Steam (MS) systems. Some of these sensing lines feed into instrument racks H22-P009 and IR-73. The instrument racks support system instruments which monitor system pressures.

The purpose of PMR/BDC 88-0254 was to protect the supports and to allow the instrument sensing lines to function when the plant is required to safely shut down during an Appendix "R" fire. The supports named in the BDC had existing carbon steel cap screws which were replaced by ones made of stainless steel. In addition, certain supports were thermolagged as required. Instrument racks H22-P009 and IR-73 had their radiant energy shield walls upgraded. This last activity was covered by BDC 55-0735-0A.

Safety Evaluation Summary

Technical Specifications no longer address fire protection concerns. The addition of thermolag and the replacement of carbon steel cap screws for stainless steel ones ensures that the supports will not fail in a design basis fire.

2.6.1.25
PMR 89-0007

This modification dealt with lowering the set point for three Emergency Core Cooling System (ECCS) relief valves slightly to assure that the pressure in High Pressure Core Spray (HPCS), Low Pressure Core Spray (LPCS) and Residual Heat Removal (RHR) systems never exceeds 110% of system design pressure as required by Section III of the ASME code. Also, relief valve springs were replaced by ones made for the new range of set pressures.

Safety Evaluation Summary

The modification did not result in a degrading of nuclear safety as described in the Licensing Basis Documents. It did, however, result in improved margins of safety for the LPCS, HPCS and RHR suction piping. Thus, there was no increase in the probability of a previously analyzed accident or the creation of the possibility of a new type of accident.

2.6.1.26
PMR 89-0103

PMR/BDC 89-0103 completed the cross tie project begun by BDC 84-1724-0D. Two butterfly valves (SW-V-933A and B) were installed and the fabrication of a removable spool piece and siphon plug were done under this design change. The normally removed spool piece, along with the valves, will allow cross-tying the two Divisions of Standby Service Water, if required, in an emergency.

Safety Evaluation Summary

This change did not increase the probability of occurrence or consequences of an accident, or malfunction of equipment important to safety because the valves will be blind flanged on one side except when they are placed in service under emergency conditions when there is a need to connect the redundant trains of Standby Service Water together to achieve and maintain Plant shutdown. A single failure of either the blind flange or the valve would not result in a pressure boundary breach. Since the cross tie is normally not in use, it does not impact the previous accident analyses for the Plant.

2.6.1.27
PMR 90-0171

The safety function of the overload relays for Containment Return Air (CRA) fans 3A,B,C and 5A,B,C,D fan motors is to passively carry the normal maximum motor load current during an accident.

This change installed overload heaters for the above-mentioned thermal overload protection relays which were one size larger than described in the Final Safety Analysis Report (FSAR). This provided sufficient margin to assure overload protection will not interfere with the safety function of the fans under accident loading with maximum or minimum design voltage at the motors.

Safety Evaluation Summary

This modification reduced the chance that maximum motor load current during an accident could cause a trip of overload protection relays and result in the loss of these fans' safety function. Therefore, this modification resulted in an increase in the level of safety for the Plant.

2.6.1.28
PMR 90-0335

The purpose of Reactor Core Isolation Cooling (RCIC) valve 8 (RCIC-V-8) is to close to provide isolation of the RCIC Steam Supply Line for a postulated line break and for containment isolation. PMR 90-0335 changed the position of RCIC-V-8 from 75% open to 65+2% open to reduce its closing time by shortening its stroke length.

Safety Evaluation Summary

This modification did not increase the probability of occurrence or consequences of an accident, or malfunction of equipment important to safety. It lessens the closure time which results in faster termination of the consequences of an accident. It results in maintaining current analysis limits related to environmental qualification and containment isolation functions. Additionally,

the RCIC system remains fully operable for its design function.

2.6.1.29
BDC 55-1420-0A

Diesel Building Mixed Air (DMA) cooling coil DMA-CC-51 was spared in place.

Safety Evaluation Summary

The cable cooling system can still maintain the prescribed ambient temperatures in the diesel generator corridors without DMA-CC-51. The change does not increase the consequences of an accident evaluated previously in the Licensing Basis Documents because the ambient temperatures in the diesel generator corridors are still below the specified temperature for which the cables are rated.

2.6.2 LIFTED LEADS AND JUMPERS (Temporary Changes)

The following are summaries of temporary changes made in the facility by use of the Lifted Lead and Jumper (LLJ) Procedure (PPM 1.3.9). As required by 10CFR50.59, each change was evaluated and determined not to represent an Unreviewed Safety Question nor require a change to the WNP-2 Technical Specifications.

2.6.2.1
LLJ 91-010, 011, 012

The Annunciator Horns for the Division I, II, and III Diesel Generators were isolated from the Class 1E Circuitry to prevent possible degradation of the diesel systems while the qualifications and isolation capabilities of the horns were verified.

Safety Evaluation Summary

This change did not result in a change to WNP-2 Technical Specifications or involve an Unreviewed Safety Question. The lifting of the leads did not impact the ability of the diesel generators to perform their design function. However, the alarms were not audible and the response to an alarm was governed by seeing the alarm window flashing on the annunciator panel rather than hearing the horn. This was overcome by assignment of two operators to the engine rooms whenever the engines were running (one to monitor the operation of the engines, and one to monitor the annunciator panel for alarms). The horns were returned to service following verification of qualification.

2.6.2.2
LLJ 91-045

Valve HPCS-V-11 was tagged out in its safety function position with the breaker open. This caused an alarm for loss of power to the valve. This alarm window also serves as the loss of power alarm for other valves. The leads for the HPCS-V-11 loss of power circuit to this alarm were lifted so that it would not mask other actual loss of power alarms which could occur.

Safety Evaluation Summary

Removal of the alarm for valve loss of power does not impact the ability to safely shut down the Plant. The valve is tagged out in the safety position. Since there is no power available to the valve, there is no need to have an alarm for loss of power to the valve. By lifting the lead, the ability to identify loss of power to other valves is restored. Loss of alarm on loss of power to HPCS-V-11 does not increase the probability of an accident since the valve is tagged out in the safety function position, closed. Also, no previously unanalyzed condition is created since the valve is in its safety position and the lifted lead simply removes an

alarm which would tell the operator that power to the valve had been lost.

2.6.2.3
LLJ 91-219

A fuse was removed to disable the circuitry that causes a load shedding of the Division 1 Service Water pump. This temporary change was made to allow the Division 1 diesel generator to remain available during the installation of a logic modification.

Safety Evaluation Summary

The fuse removal described above was made to the Division 1 logic. This occurred during Refueling Outage R6. Division 1 was not required to be operable during the time the fuse was removed. Removal of the fuse allowed the Division to remain in service, providing greater operations flexibility and an increased level of redundancy over that assumed for the shutdown condition when only one Division of emergency power is required. Removal of the fuse does not create the possibility of a new type of accident in the shutdown condition. The probability of an accident was not increased since only one Division was assumed to be available in past analyses. This change did not result in the creation of an unreviewed safety question.

2.6.3 FSAR CHANGES

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

2.6.3.1
SCN 91-075, Minor Cycle 7 Reload Changes

The purpose of this change was to accurately describe the reactor core as loaded for Cycle 7. Certain minor changes to the FSAR were required.

These changes included the first use of 9X9-9+ fuel assemblies along with the results of the analyses performed for the reload.

Safety Evaluation Summary

No change to the Technical Specifications was required and no Unreviewed Safety Question resulted from these changes because the analyses, in all cases, demonstrated that core transient and steady state limits would not be significantly affected. WNP-2 now has a COLR in lieu of cycle specific fuel limits in the Technical Specifications.

2.6.3.2
SCN 91-076, Nuclear Fuel Channel Program Transition

This SCN documents the plans to transition from the current practice of using nuclear fuel channels for longer than one fuel assembly lifetime to use of the channels for a single assembly lifetime. Once this transition is complete, it will eliminate the need to perform a detailed channel inspection.

Safety Evaluation Summary

This change does not involve an unreviewed safety question since the nuclear fuel channels will now be used for a smaller exposure period than was previously the case. This will greater reduce any risk of channel box deflection. Therefore, the need for channel inspections, other than to support their reuse during the transition period, will not be required.

2.6.3.3
SCN 91-043, FSAR Radiation Level Drawing Changes

This FSAR change involved updates to the drawings showings expected radiation levels in the Plant. These changes to the drawings were required due to the increase in background radiation levels in several area within the Plant.

Safety Evaluation Summary

The Safety Evaluation concluded that the changes made to the drawings did not involve an unreviewed safety question since the increased radiation levels are a result of normal Plant operations and do not impact equipment qualifications. The ALARA program remains an important part of the Plant radiation protection goals. The increased levels are not high enough to significantly impact personnel movements throughout the Plant, and does not impact equipment qualification.

2.6.3.4
SCN 91-054, Emergency Plan Change

This evaluation was performed for the removal of the Yakima Valley Memorial Hospital from the Emergency Plan.

Safety Evaluation Summary

The removal of the Yakima Valley Memorial Hospital from the Plan does not impact the ability to effectively implement the Plan. There are three hospitals in the immediate area of the Plant that serve as emergency health care facilities. These three facilities exceed the two recommended in NUREG-0654 and are capable of providing the anticipated required services.

2.6.3.5
SCN 91-056, Emergency Plan Change Regarding Field Team Deployment

The field team deployment time goal was changed from 30 minutes during normal working hours to 30 minutes after the team arrives at their emergency duty station.

Safety Evaluation Summary

The field teams are dispatched into the field to provide radiological data to back up information available from the Plant monitoring systems. The information gathered is part of the information used in the determination of what Protective Action Recommendations are to be made. These teams will still be sent into the field in an expeditious manner. The teams will thus still get to the field sooner during normal working hours when personnel are already on-site. Since the information obtained is a backup for information obtained from installed instrumentation, and since the identified change does not reflect a change to the manner in which the Emergency Plan is implemented, it was determined that this change does not adversely impact Plant or public safety.

2.6.4 PROBLEM EVALUATIONS

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

2.6.4.1
PER 91-0016

A High Pressure Core Spray (HPCS) Diesel Generator outside/recirculation air damper was found stuck in the closed position due to a broken positioning shaft. The damper was left in the closed position. Replacement parts were not readily available.

Safety Evaluation Summary

This damper is used to admit outside air into the diesel room. It is not used for diesel air intake. The damper can be manually positioned. Procedures were deviated to provide directions for when, based on outside temperatures, the damper needs to be open. There is adequate time for this manual repositioning even in an emergency condition. The HPCS diesel, with the damper opened manually when needed, will continue to operate as originally designed. Therefore, operation with manual repositioning of the air damper when needed does not constitute an unreviewed safety question. This condition was evaluated as temporary until the damper could be fixed. It was not intended as a permanent change.

2.6.4.2
PER 291-0022

The annunciator horns at the local emergency diesel generator panels are powered from Class 1E power sources but are not qualified to Class 1E, Seismic I requirements.

Safety Evaluation Summary

The horns at the local panels were disconnected until qualified separation could be provided or the horns could be qualified. This does not impact the local annunciators or indications or the annunciators in the Control Room. A second individual will be present at the diesels during testing to help monitor the engine conditions until the horn can be placed back in service. The diesel generators are designed to start and run without anyone present. This is accomplished either automatically or manually

from the Control Room. The flashing of the local control panel annunciators, and the Control Room annunciators flashing and alarming, are not affected by this occurrence. Therefore, the margin of safety as defined in the licensing basis documents is not impacted by this change, no previously analyzed accident frequencies are increased, and no new accident scenarios are created.

2.6.4.3
PER 91-069

Regulating transformer RPS-TR-6BER was not outputting the desired voltage. A change was made to jumper around the transformer.

Safety Evaluation Summary

This regulating transformer is part of the alternate power supply to the Reactor Protection System (RPS) busses. This alternate power supply is non-safety-related, QC-2, and is not credited in the accident analyses for the Plant. The Class 1 RPS is protected from this power supply by safety-related electrical protection assemblies which trip breakers on overvoltage, undervoltage or underfrequency. Jumpering out the regulating transformer maintains the alternate power supply in service without effecting the electrical protection assemblies. In addition, the normal power supplies, the RPS motor-generator sets, are not impacted by this change. None of the power supplies are required for the RPS to perform its intended safety function. The RPS is a normally energized logic that performs its functions on de-energization caused by either logic change or loss of system power. The change described in this PER does not result in a new or different type of accident. Since the probability or consequences of an accident already analyzed is not increased by this change, no unreviewed safety question was involved.

2.6.4.4
PER 91-0078

During a snubber optimization reanalysis for main steam loops B and C it was determined that potentially nonconservative steam hammer reactive loads existed at five pipe supports.

Safety Evaluation Summary

These findings were preliminary in nature. Additional analyses were required to validate the results. A 50.59 review was performed to justify continued operation until the results could be validated. In all cases, the pressure boundary pipe stresses are within the ASME Code limits if stress allowables are developed from actual material test data. Flange loading of the Main Steam Relief Valves in all cases remain far below the manufacturers allowable limits ensuring relief valve operability.

Therefore, even if the preliminary analyses proved to be correct, the piping still meets the original conditions assumed in the accident analyses and does not result in a reduction in the assumed margin or in the potential for a previously unanalyzed accident.

2.6.4.5
PER 91-0169

This PER documented failure to meet Regulatory Guide 1.75 separation criteria for MSIV Leakage Control (MSLC) system and Reactor Recirculation (RRC) system valves. This problem was found during a single failure review with the PER dated March 8, 1991.

Safety Evaluation Summary

Each of the circuits in questions are control circuits and are therefore considered low energy. Each of the circuits are provided with a circuit protection device to clear faults which could occur. In the case of the MSLC system, the circuits are not required to be protected under Regulatory Guide 1.75 criteria, but were included in the evaluation as a personnel safety matter due to the potential for a radiation release inside the Reactor Building should a fault impacting both Divisions occur simultaneously. The probability of a fault impacting redundant trains coincident with a design basis accident is very small. This would require a fire in the immediate area of the cables. As a compensatory measure, a fire tour was ordered for the areas in question. The RRC separation problems were corrected during Refueling Outage R6 which began April 12, 1991. The MSLC valves in question are normally closed, are required to be closed for containment isolation, and are only required to open post-LOCA for off-site dose considerations. Recent generic analyses have shown that the MSLC system may not be important to the control of off-site releases.

The probability of an accident was not increased by these separation concerns since separation issues are not initiating events. Operation with the separation issues described above does not increase the probability of an accident from that previously evaluated due to the extremely low probability of a fault resulting in loss of both divisions on protected control circuits such as these.

2.6.4.6
PER 91-0391

It was determined that several Technical Specification required radioactive liquid and gaseous effluent monitors do not include the trip and/or alarm functions required to be verified as part of the channel functional test (CFT), or the trip and/or alarm

functions were not being tested as part of the CFT. This condition was reported in LER 91-013-02. The Plant was in a Cold Shutdown condition when this problem was found.

Safety Evaluation Summary

The monitors in question were all capable of performing their intended safety functions. Each of the monitors was operating as designed and met the design basis requirements. The concern was that, because of the details of the equipment design or function, there was not always a technical need to provide the trip (indication only for some monitors) or alarm (no "controls not set in operate mode" alarm since the monitor cannot be set in anything other than the operate mode) functions called by the Technical Specifications for testing. Since it is not possible to test some monitors in the manner called for by the Technical Specification, and since the condition being tested for cannot occur, it was concluded that the Technical Specification requirements are satisfied to the maximum extent possible by the testing being performed. The appropriate Action Statements were followed until a Technical Specification amendment was obtained for those monitors where the full Technical Specification required testing could not be met . Since the design basis for the monitors, the Technical Specification requirements, and the monitoring functions are being met, it was concluded that the no increase in the probability of an accident previously analyzed or possibility of an accident not previously analyzed is involved. No Technical Specification change was required since the appropriate action statement requirements were met.

2.6.4.7
PER 290-0567

PER 290-0567 was written to correct problems with fuel oil storage tank level switches. Specifically, the PER documented concerns involving consistency in the instrument outputs and the potential impact of these inaccuracies.

The immediate corrective action was to declare the Diesel Oil level switches, DO-LITS-10A and 10B, inoperable. Plant procedures were revised to require all Technical Specification surveillance measurements of the quantity of fuel in the tanks to be made by manual dipsticking.

Safety Evaluation Summary

The inaccuracy of the diesel fuel storage tank level instruments did not increase the consequences of any accident or transient described in the Licensing Basis Documents because the inaccuracy of these instruments did not prevent or degrade any actions assumed in previous analysis. The diesel fuel storage tank level instruments are passive in that they have no function in the fuel

oil system operation. Therefore, they could not have prevented the starting and running of the diesel generators. Their function is to provide information on the quantity of remaining oil. This information is also available from manually dip-sticking the tanks.

2.6.4.8
PER 91-0643

Re-calculation of off-site dose consequences for design basis accidents was performed due to new meteorology data. It was determined that five outside containment isolation valves located between inboard and outboard Main Steam Isolation Valves must be maintained closed when the Plant is at power to limit the postulated off-site dose.

Safety Evaluation Summary

An evaluation was performed to determine the acceptability of operating with the five valves, which serve as drain valves off the main steam lines, closed. The concern was the potential thermal cycle effects on these valves and their associated piping from opening the valves at power. A review of past analyses showed that the worst case usage factor for these valves was 50 cycles of the type envisioned. Administrative controls were provided to maintain the valves in the closed position at power and to limit the number of times the valves are opened at power. These controls ensure that the valves are cycled only a fraction of the number of allowable cycles. Therefore, the Plant remains well within the design basis, the probability of a previously analyzed accident is not increased, and the possibility of a new type of accident is not created.

2.6.4.9
PER 91-0656

New setpoint calculations found that the time delay after diesel generator start until start of the RHR and LPCS pumps should be increased.

Safety Evaluation Summary

The new setpoints assure that the bus voltage is sufficient to support pump start. The longest time allowable is determined from the accident analyses and the time period assumed for the pumps to come up to the required accident flow. In all instances the new relay setpoints are within the allowable time period

assumed in the accident analyses. Since the new setpoints are within the design basis assumed values, and since they provide a more accurate setting, the probability or consequences of a previously analyzed accident is not increased and no new accident type is created.

2.6.4.10
PER 91-0728

The moisture resistant cover of a flexible conduit was found to be damaged. A weep hole was drilled in the bottom of the junction box that the conduit is attached to as an interim corrective measure. Replacement of the conduit cover at power is not possible.

Safety Evaluation Summary

The weep hole allows any moisture which might collect in the flexible conduit after an accident cover to leak out of the junction box. There are no conduits exiting from the bottom of the junction box. The equipment in the junction box has been qualified for the post LOCA environment. Past testing has proven the effectiveness of the weep hole for condensation removal without negative impacts. Since weep holes are a previously tested configuration which effectively removes moisture buildup in the box, and since the equipment within the box is qualified, this change does not increase the probability or the consequences of a previously analyzed accident or create the probability of an accident no previously analyzed.

2.6.4.11
PER 91-0956

Insulation damage was found on a 12 inch heater drain line that is indicative of a water/steam hammer event. A NDE inspection of the pipe revealed no structural damage. The piping was restored to its pre-event condition. However, a 50.59 evaluation was performed to justify continued operation until modifications could be made to preclude potential future events.

Safety Evaluation Summary

A 50.59 evaluation was performed using the worst case assumption that the line in question would experience a doubled ended break. This section of piping is not credited in the Final Safety Evaluation Report since it is downstream of the MSIVs. Failure of this line would result in a less severe transient than the previously analyzed main steam line break. Therefore, continued operation could not result in a different type accident than previously evaluated. Operation with the pipe in the pre-event condition does not result in an increase in the probability of an accident since, based on engineering judgement, the pipe is as structurally sound as before the event except for the addition of

one thermal cycle. Piping in the Heater Drain system is designed to withstand multiple thermal cycles.

2.6.4.12
PER 91-0971

As identified and reported in LER 91-029 the Containment Atmospheric Control (CAC) system hydrogen recombiners would not work properly in the automatic mode. Procedures were changed to require manual operation of the system.

Safety Evaluation Summary

The CAC system is required to be operated in a hydrogen generation event. The hydrogen level will not reach a condition where the CAC system is needed for at least six hours post-accident. This provides adequate time to bring an additional individual on site to operate the system. It was always possible to operate the CAC system manually. Operation in the manual mode does not increase the probability of an accident previously analyzed or result in an accident not previously analyzed.

2.6.4.13
PER 91-0973

In November 1991 concerns were expressed about the functionality of the tornado latches for the Reactor Building crane.

Safety Evaluation Summary

The Reactor Building crane is provided with tornado latches to prevent crane movement and potential damage to safety related equipment due to a tornado. These latches were found in a condition which called into question their ability to perform. A 50.59 review was performed to support continued Plant operation for an interim period of time until the latches were fixed. Several restrictions were applied, including: 1) the crane was to remain in the parked position except to test the latches; and 2) the Plant was to begin a controlled shutdown if a tornado warning were issued for the Hanford area. The probability of a tornado at WNP-2 is $6X10^{-6}$. The recorded tornadoes are seasonally dependant, with all recorded tornadoes falling in the March-September time frame. The probability of a tornado hitting WNP-2 in November is thus extremely small. Operation with the Reactor Building crane in the parked position with the tornado latches not functional for a short period of time in November does not increase the probability of an accident previously analyzed nor create the possibility of an accident not previously analyzed.

2.6.4.14
PER 91-1009

Information was received from the SOR company that pressure switches containing a Kapton diaphragm, when placed in service in a water environment, may fail prior to the 40 year expected life. This PER documenting the applicability to WNP-2 was written on December 16, 1991.

Safety Evaluation Summary

WNP-2 currently has 25 of the subject pressure switches in service in water environments. A 50.59 evaluation was performed to justify continued operation through the next refueling outage without a documented qualified life for these switches. The next refueling outage is scheduled to begin in April 1992.

The Kapton material, when exposed to water, degrades to a given level and then stops deteriorating. Failure at that point is not imminent. At that point failure may occur if a system pressure transient occurs. The two documented failures of these pressure switches at WNP-2 occurred as a result of water hammer events. The systems affected have been modified to preclude recurrence. Since the switches have not exhibited failure characteristics except during water hammer events, and since water hammer events are infrequent, this is not considered a common mode failure concern for the short time period covered by this 50.59.

Continued operation for the remainder of the fuel cycle with these switches does not increase the probability of an accident since switch failure by itself would not result in an accident, either analyzed or not previously analyzed.

2.6.5 PLANT TESTS AND EXPERIMENTS

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

There were no Tests or Experiments performed under the provisions of 10CFR50.59 in 1991.

2.6.6 PLANT PROCEDURE CHANGES

The Plant Procedure control program requires a 10CFR50.59 evaluation whenever a procedure is changed. This provides assurance that the change does not require a change to the Technical Specifications or involve an Unreviewed Safety Question. Plant procedure changes associated with other change documents such as Plant Modifications or Lifted Leads and Jumpers are described in other subsections of this report. The following are summaries of significant Plant Procedure changes not covered elsewhere in this report that were processed during 1991.

2.6.6.1
Procedure Deviation Form Number 91-698, 91-708, 91-709, 91-717, 91-884, 91-1044, 91-1045 and 91-1064

The monthly and semi-annual operability tests for the High Pressure Core Spray (HPCS) system were deviated due to the inoperability of the exhaust fan for the HPCS diesel fuel oil transfer pump room. The associated safety evaluation included an analysis for continued HPCS operability with the room fan inoperable.

Safety Evaluation Summary

The room exhaust fan is not required for transfer pump operability. The worst case postulated room temperatures do not exceed the pump operational temperature rating. The fan was installed to remove fumes from the room for personnel safety considerations. Personnel are not required to enter the room post-accident. Compensatory measures were implemented to exhaust the fumes and to monitor the room conditions during the worst case fume accumulation conditions. Worst case fume conditions occur when an auxiliary steam line routed through the room is placed in service to support nitrogen storage tank operation for containment inerting.

The fan is not required to ensure the equipment in the room remains operable and is capable of performing its intended safety function. The lack of the fan does not result in the creation of a new type of accident. Compensatory measures have been taken. There is no unreviewed safety question as a result of these procedure deviations.

2.6.6.2
Procedure Deviation Form Number 91-646

Valve HPCS-V-4, the HPCS injection valve, is interlocked to prevent opening when the reactor pressure vessel level is above Level 8. This procedure deviation provided a method to allow testing of the valve when the Plant is in the Cold Shutdown and

Refueling modes of operation.

Safety Evaluation Summary

The HPCS systems primary function is to provide a high pressure safety grade source of water for reactor cooling post accident when the reactor remains pressurized. This procedure deviation provided a method to support testing of HPCS-V-4 with the Plant in the Cold Shutdown or Refueling modes by defeating the reactor high Level 8 interlock. This is only allowed when the HPCS system is not required to be operable and the HPCS pump control fuses are removed to prevent the pump from starting. This serves the same function as the interlock in that no high pressure water from the HPCS system can be injected into the reactor pressure vessel with the vessel level high, thus preventing an overpressurization event.

The controls provided in the procedure to ensure HPCS cannot inject water for the short period of time HPCS-V-4 is open for testing provides a level of protection equivalent to that which would be provided by the interlock. The administrative control provided by the procedure ensures the protection is maintained for the duration of the testing. Since an equal level of protection is provided by the procedure, the probability of an accident is not increased. The controls provided also ensure that no new type of accident not previously analyzed is introduced.

2.6.6.3
Procedure Deviation Form Number 91-879, 91-880, 91-885, and 91-896

As a result of new analysis for the Reactor Building and Standby Gas Treatment (SGT) system performance during unusual weather conditions, it was necessary to reduce the control setpoint for the SGT system to maintain the building at a slightly greater vacuum condition.

Safety Evaluation Summary

The setpoint change resulted in a slightly greater vacuum condition in the Reactor Building. All of the equipment in the Reactor Building is designed to operate at a vacuum condition greater (more negative) than that created by this change. The only real impact of this change is to personnel egress through the building doors. The higher vacuum may result in a small number of personnel having difficulty opening these doors. This is not a significant impact since the Operations personnel carry radios and can call for help if required. Other personnel can use the Plant page or phone systems to request help in opening the doors. It is expected that only a small number of personnel may be impacted by this change.

This change does not increase the probability of an accident since system pressure is not an initiating event. Since all personnel will still have access to the building, either alone or with assistance, this change does not result in the creation of a new type of accident.

2.6.6.4
Procedure Deviation Form Number 91-224, 91-227, and 91-322

Procedures were deviated to require the declaration of an Site Area Emergency for an "Operating Basis Earthquake" (OBE) instead of a "Safe Shutdown Earthquake" (SSE).

Safety Evaluation Summary

The SSE level can only be determined to have occurred by sending tape records of seismic event data off-site for analysis. The OBE level earthquake can be determined on-site. In addition, the OBE is a lower magnitude earthquake than the SSE. Therefore, this change is in the conservative direction in that a Site Area Emergency will be declared at a lower earthquake level. Therefore, this change is in the conservative direction and does not involve an Unreviewed Safety Question.

2.6.6.5
Procedure Revision Form for PPMs 5.5.5, 4.12.4.1, and 4.12.4.1A

PPM 5.5.5, Overriding RCIC Low RPV Pressure Isolation Interlock, an Emergency Operating Procedure, was changed to meet the Emergency Procedure Guidelines, Revision 4. One exception to Revision 4 was taken. The RCIC steam supply system low pressure isolation logic is bypassed in the procedure to allow continued operation of the RCIC system. Operation at low steam pressure does not result in damage to the turbine. It does, however, provide continued flow to the core to help maintain core coverage. The system isolation for mitigation of the analyzed high energy line breaks is not impacted.

Procedure, PPM 4.12.4.1, Fire, was revised to meet the Emergency Procedures Guidelines, Revision 4.

PPM 4.12.4.1A, High Energy Line Break, was developed to provide guidance to Plant personnel in the event of a high energy line break.

Safety Evaluation Summary

The changes to these procedure were made to bring the procedure into close alignment with the NRC approved Emergency Procedure Guidelines, Revision 4. Exceptions were taken to ensure the changes did not result in the Plant being outside the design or licensing bases. The changes do not result in the Plant being in

a condition not previously analyzed or increase the probability of an accident previously reviewed. The changes are in accordance with the Revision 4 guidelines and therefore constitute changes previously reviewed by the NRC. Therefore, these changes do not constitute an unreviewed safety question.

2.6.6.6
Procedure Revision Form for PPM 9.5.5

Problems were experienced with control rod multiple notching during rod withdrawal. The procedure was revised to allow an increase of the rod withdrawal/insertion time (a decrease of the withdrawal speed) to greater than 60 seconds.

Safety Evaluation Summary

The control rod withdrawel speed does not affect the ability, or the time required, to scram a control rod. As described in the FSAR, the only restriction on control rod speed is that the withdrawal time shall be greater than 40 seconds to limit the effects on the fuel as the rod is withdrawn.

This procedure change is in the conservative direction. This change does not result in a condition not previously analyzed or cause an increase in the probability of an accident previously analyzed. Therefore, this change does not constitute an unreviewed safety question.

2.6.6.7
Procedure Revision Form for PPM 2.5.7

The main turbine generator procedure was changed to permit bypassing the main turbine first stage pressure switches for performance of the bypass and throttle valve testing below 25% of rated thermal power with the main generator not synchronized to the grid.

Safety Evaluation Summary

The main turbine first stage pressure switches provide a signal to the Reactor Protection System (RPS) logic to block a reactor trip due to turbine throttle or governor valve closure, or governor valve trip system low oil pressure, when first stage pressure is less than 165 psig. The 165 psig is the pressure equivalent to 30% rated thermal power.

This procedure change allows isolation of the pressure switches for valve testing when the Plant is below 25% rated thermal power and the generator is not synchronized to the grid. This results in the RPS trips due to turbine throttle or governor valve closure, or governor valve trip system low oil pressure, remaining bypassed as required by Technical Specification Table

3.3.1-1 when the Plant is below 25% power. This is necessary due the minor first stage pressure spikes that can occur during valve testing.

This procedure change provides added assurance that valve testing below 25% rated thermal power will not result in an unnecessary reactor trip due to minor pressure spikes. This change is in accordance with Technical Specification Table 3.3.1-1 requirements. This change is also consistent with the the FSAR system description in that the pressure switch isolation provides assurance that the valve trip functions are bypassed below 30% of rated thermal power as designed.

This procedure change provides assurance that the Plant is operated per the design. Therefore, this change does not create the possibility of an accident not previously analyzed or increase the probability of accident previously analyzed. This change does not result in an unreviewed safety question.

2.6.6.8
Procedure Revision Form for PPM 6.3.23

Plant procedure PPM 6.3.23, Revision 0, "Handling Irradiated Fuel in the Spent Fuel Pool" was developed to ensure irradiated fuel is transported at the proper depth in the spent fuel pool.

FSAR section 9.1.4.3 states that the Material Transport System jib cranes (MT-CRA-9A and 9B) are designed and qualified to handle new and spent fuel and other components in the work area. The FSAR also describes an adjustable travel limit switch designed to stop the hoist from raising the spent fuel above approximately 8 feet below pool level. The jib cranes do not have this limit switch.

This new procedure accomplished the intent of the FSAR statements in regard to the required distance the active fuel should be kept below the surface of the spent fuel pool. Specifically, MT-CRA-9A and 9B now require a lifting sling such that when the jib crane block is fully raised the top of the active fuel will remain at least 8 feet below the pool surface. There were no changes to the operation of the refueling platform.

Safety Evaluation Summary

Fuel handling accidents involving the jib cranes and the refueling platform were previously evaluated in the Licensing Basis Documents. This procedure revision did not change fuel handling activities other than the addition of the lifting sling. This change did not increase the probability of occurrence of a malfunction of equipment or the consequences of the malfunction of equipment important to safety. The use of the sling makes it physically impossible to lift the fuel too high.

2.6.7 FIRE PROTECTION PROGRAM CHANGES

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

SCN 91-015

Section F.3 of the WNP-2 FSAR Fire Protection Evaluation, contains a comparison of the plant fire protection program against the guidelines of BTP APCSB 9.5-1. FSAR page F.3-52 (Section D.2.c, Control of Combustibles) states that although the use of PVC cabling is minimized, halogenated plastics have been used in wiring for lighting. SCN 91-015 revised this section to clarify that PVC had been used in both lighting and communications cabling installed in the Plant, and that this wiring is primarily installed in conduit. This change documents the existing Plant configuration - no new cabling is associated with this change.

Safety Evaluation Summary

The safety evaluation for the SCN states that the existing Plant configuration does not represent an unreviewed safety question as:

- the cabling is appropriately qualified for the installed environment
- the cabling is primarily routed in conduit which will serve to isolate the cabling from ignition, as well as to minimize the effects of chlorides which could be produced by combustion.
- the fire loading due to the installed cabling has been previously evaluated in the FSAR fire hazards analysis, and
- the use of halogenated plastics (such as PVC cable jacketing) within the Plant has been previously evaluated.

SCN 91-017

This change modifies the description of the WNP-2 fire protection program to incorporate existing procedural deviations which state that fire protection system testing required by FSAR Appendix F.5 (Essential Fire Protection System Testing/Operability Requirements) or plant commitments to NFPA Standards or the fire insurance underwriter, may not be performed during power operation if that testing would require entry into high radiation areas.

Safety Evaluation Summary

The safety evaluation for the SCN states that the incorporation of the changes in fire protection system testing frequencies do not represent an unreviewed safety question as:

- the change does not affect essential (formerly Technical Specification) fire protection water spray, sprinkler, or halon systems, or fire hose stations, as these systems are not installed in high radiation areas.
- the change could affect the six month visual surveillance of certain fire detection instruments, but the yearly functional test of these instruments is not changed and provides assurance of operation.
- the change could also affect the surveillance of the second side of certain fire doors and penetration seals installed in high radiation boundaries. Since work in high radiation areas is strictly controlled, these barriers are not subject to inadvertent damage. Inspection from one side provides reasonable assurance that the barriers will continue to perform their intended function.
- previous testing of the effected systems has not indicated any recurring failure which would warrant regular entry into high radiation areas for the purpose of performing this testing.

One additional change is the transformer retrofill. The transformers are being retrofilled with a non-PCB based transformer fluid. The fluid retrofill is being performed under Substitution Evaluation No. 292. The 50.59 review for the substitution evaluation did not identify any safety issues. However, certain fire protection related issues were identified and resolved under PER 289-0274. The transformer retrofill in the Turbine Building and in the Division 2 Switchgear Room were completed during the 1991.

The evaluation of the PER issues recommended the upgrade of certain penetrations and dampers in the transformer rooms. These modifications are complete. The FSAR changes associated with the retrofill work completed to date are documented in SCN 291-091.

2.7 REPORT OF DIESEL GENERATOR FAILURES

This section contains information regarding diesel generator failures, valid and nonvalid, in accordance with the requirements of WNP-2 Technical Specification 4.8.1.1.3. WNP-2 experienced a total of one valid failure and no invalid failures in 1991 for the three emergency diesel generator units.

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

   Division One Emergency Diesel Generator (DG-1)
   August 1, 1991

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

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

3. Cause of failure:

   During performance of the Technical Specification required monthly surveillance test, the Division One Diesel Generator exhibited unexpected frequency oscillations. The diesel generator output breaker was tripped and the engine speed was reduced to the slow speed setting where the oscillations did not occur.

4. Corrective measures taken:

   The diesel engine governor is equipped with a magnetic pickup signal selector (MPPS) module which receives a signal from the magnetic pickup transducer (MPU) located on each engine near the flywheel. Each of the Division 1 and 2 diesel generator units is made of two engines connected to a single generator. There is a MPU for each of the engines. The MPU puts out a frequency signal. The single MPPS module chooses the higher frequency from the two MPU signals it receives and controls engine speed based on this input. The MPPS module is phase sensitive, so that if the two MPU signals are nearly in phase, the MPPS module has difficulty distinguishing between the signals and can switch control back and forth between the signals. This resulted in the speed oscillations observed. The leads on one of the MPUs were swapped which resulted in the signals being far enough out of phase from the second MPU signal that the MPPS module could distinguish between them. The speed oscillations were eliminated.

5. Length of time diesel generator unit was unavailable:

   The Diesel Generator was out of service for 15 1/2 hours, and returned to service at 1810 Hrs on Sunday January 7, 1991. The Plant was in a Cold Shutdown condition during the time of this diesel outage.

6. Current surveillance test interval:

   Thirty-one days.

7. Verification of test interval:

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

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GL Gelhaus	PE27	CH McGilton	PE21	MP Reis	PE18
GJ Godfrey	1023	SL McKay	9270	GC Sorensen	280
LL Grumme	280	DL Larkin	PE23	DA Swank	PE18
JD Harmon	927S	AL Oxsen	387	GA Tupper	325
LT Harrold	927M	JV Parrish	1023	RL Webring	PE27
AG Hosler	PE21	DJ Pisarcik	927K	Docket File	
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Docket No. 50-397

February 28, 1992
G02-92-054

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Martin:

SUBJECT: **NUCLEAR PLANT WNP-2, ANNUAL OPERATING REPORT 1991**

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

2) WNP-2 Technical Specifications, 6.9.1.4 and 6.9.1.5

3) Regulatory Guide 1.16, Reporting of Operation Information Appendix A

In accordance with the above listed references, the Supply System hereby submits the Annual Operating Report for calendar year 1991. Should you have any questions or comments, please contact G. L. Gelhaus, WNP-2 Assistant Plant Technical Manager.

Very truly yours,

J. W. Baker
WNP-2 Plant Manager

JWB:DAS:KEP:cgek
Attachments

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

Authors: DA Swank/KE Pisarcik			For Signature of: JW Baker		
ion:					
For Approval of:	MP Reis	RL Webring	JD Harmon	SL McKay	DJ Pisarcik
Approved:					
Date:	2/26/92	2/27/92	2/26/92	2/26/92	2-28-92