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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
P.O. Box 968 • Richland, Washington 99352-0968

February 28, 1997
G02-97-043

Docket No. 50-397

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

Gentlemen:

Subject: **WNP-2, OPERATING LICENSE NPF-21**
ANNUAL OPERATING REPORT 1996

References:
1) Title 10, Code of Federal Regulations, Part 50.59(b)
2) WNP-2 Technical Specifications 4.8.1.1.3, 6.9.1.4 and 6.9.1.5
3) Regulatory Guide 1.16, Reporting of Operating Information, Appendix A
4) NEI Guideline for Managing NRC Commitments

In accordance with the references, the Supply System hereby submits the annual operating report for calendar year 1996. If you have any questions or desire additional information pertaining to this report, please contact Ms. Lourdes Fernandez at (509) 377-4147.

Respectfully,

R.L. Webring
Vice President, Operations Support/PIO
Mail Drop PE08

IE47 1/1

JDA/lm

Attachment 100034

cc: EW Merschoff, NRC - Region IV
KE Perkins, NRC - Region IV, WCFO
TG Colburn, NRC - NRR
REIRS Project Manager, NRC - NRR
NRC Resident Inspector (MD 927N)
DL Williams - BPA (MD 399)
NS Reynolds - Winston & Strawn

9703100241

WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1996

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

Washington Public Power Supply System
P.O. Box 968
Richland, Washington 99352

TABLE OF CONTENTS

1.0 INTRODUCTION

The 1996 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. The plant is a 3486 MWt, BWR-5, which began commercial operation on December 13, 1984.

On March 2, 1996 WNP-2 had operated continuously for 242 consecutive days when the plant was taken off line at the request of the Bonneville Power Administration, customer for WNP-2 electricity. The 242 consecutive day run was just 15 short of the plant's record mark of 257 days, set between August 1993 and April 1994. This also represented the first breaker-to-breaker run.

The plant was maintained in this reserve shutdown condition for more than one month before the refueling outage scheduled start date due to an abundance of relatively inexpensive power from the Federal Columbia River Power System.

On April 13, 1996 the plant officially entered the 1996 Maintenance and Refueling Outage (R-11) as scheduled. The plant ended the annual outage on June 21, 1996. Following startup, the plant was manually scrammed from 28.5 percent power on June 24, 1996 due to an unexpected plant response during testing of the recently-installed Digital Feedwater Level Control System. The unexpected response was due to a digital feedwater controller error. A change was made to the controller software and plant startup resumed.

From June 1996 through August 1996, power production was limited due to problems associated with the recently-installed Reactor Recirculation System Pump Adjustable Speed Drive and Digital Feedwater Level Control Systems. Power production continued to be periodically limited during the remainder of the year due to problems with the adjustable speed drives.

The Bonneville Power Administration, due to abnormally high run-off conditions on several occasions throughout the remainder of the year, also requested that WNP-2 reduce power levels so that the federal power marketing agency could maximize its generating capability from the region's hydroelectric projects.

The eleventh refueling outage was successfully completed during 1996. Significant planned and emergent activities included:

- Installation of an Adjustable Speed Drive System for the Reactor Recirculation System pump motors.
- Installation of a Digital Feedwater Level Control System.

- Installation of clamps on each of the jet pump sensing lines. The 80 new clamps (four on each line) supplement welded supports and were installed in support of the Adjustable Speed Drive System modification.

- Full core off-load to support the jet pump sensing line modification.

- Core refuel with Asea Brown-Boveri (ABB) assemblies. This represented ABB's first refuel load for a U.S. commercial nuclear power plant.

- Visual inspection of the Reactor Pressure Vessel.

- Cleaning of Main Condenser tubes.

- Refurbishment of 20 Control Rod Drive Mechanisms.

- Inspection of the Moisture Separator Reheater

During December 1996, a new power generation record was set. The gross generation for the month was 882,470 megawatt-hours and net generation equalled 850,855 megawatt-hours. The previous record of 868,390 megawatt-hours gross and 837,936 megawatt-hours net occurred in October 1995.

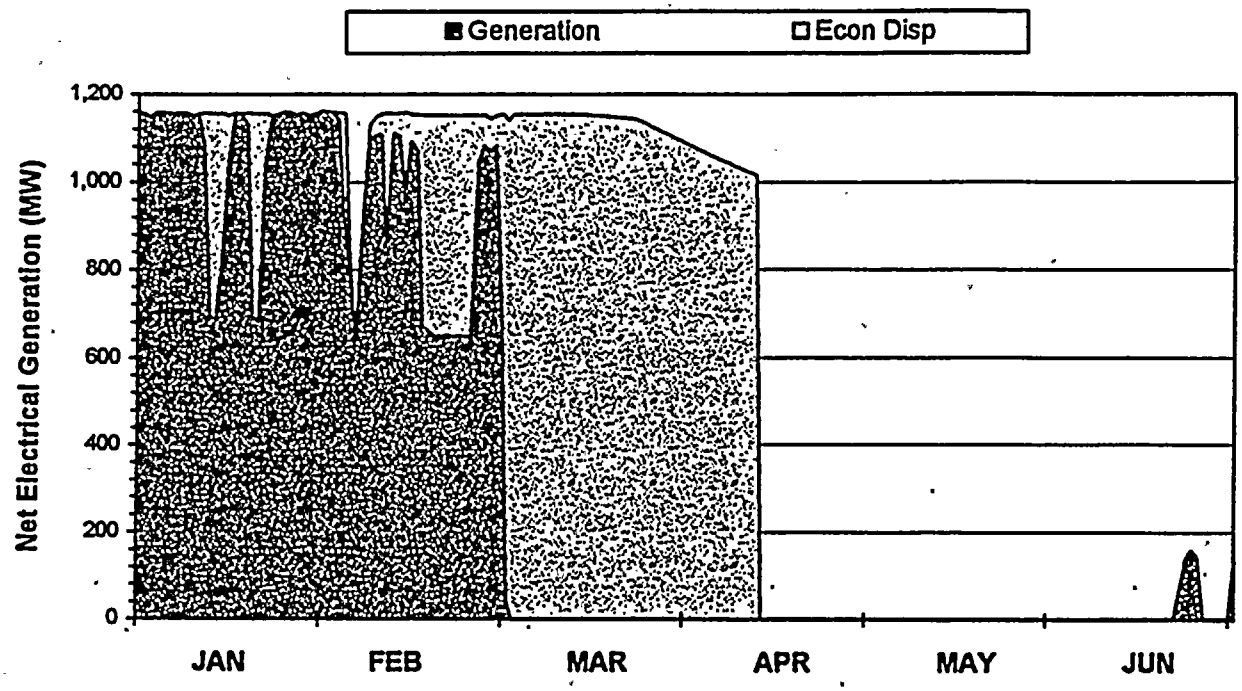
1.1 Capacity Factors - 1996

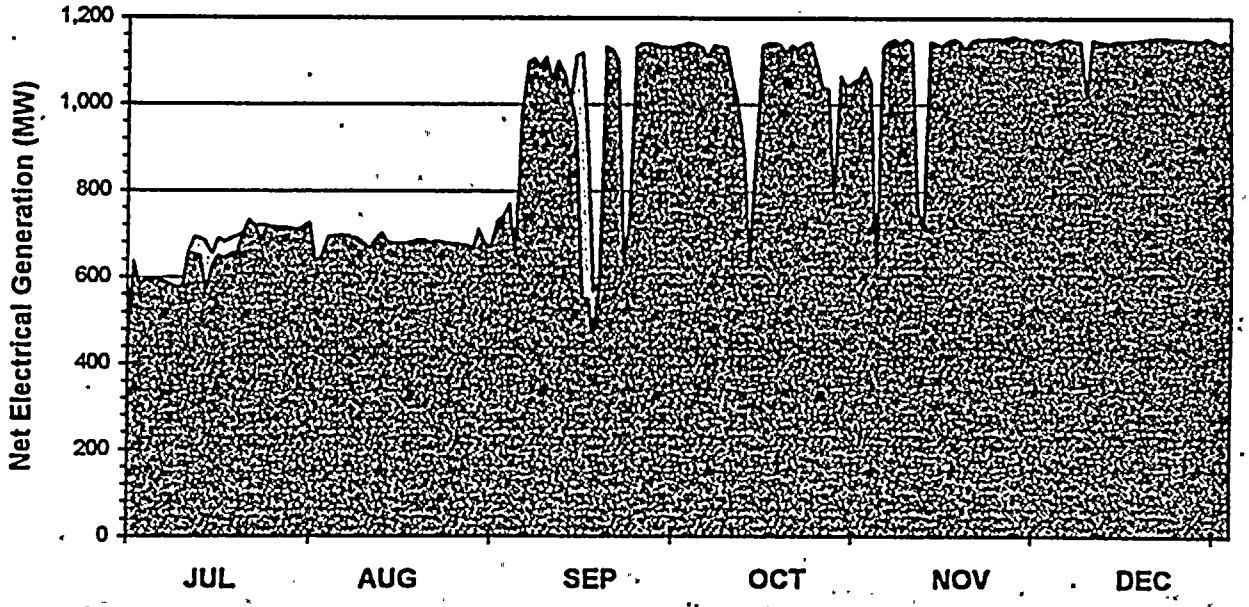
The 1996 capacity factors, based on net electrical energy output are listed below.

Month	Capacity Factor
January	97.8
February	82.1
March*	1.1
April**	0
May	0
June***	0.8
July	58.6
August	61.6
September	86.5
October	96.2
November	99.3
December	103.3
Overall	57.1

* Entered Economic Dispatch Reserve Shutdown Condition
** Started Maintenance and Refueling Outage
*** Ended Maintenance and Refueling Outage

1.2 Load Profile - 1996





2.0 REPORTS

The reports in this section are provided pursuant to: 1) the requirements of Technical Specifications 6.9.1.4 and 6.9.1.5, "Annual Reports," 2) the requirements of Technical Specification 4.8.1.1.3, "Reports" (Electrical Power System Surveillance Requirements), 3) the requirements of 10CFR50.59, "Changes, Tests, and Experiments," 4) the guidance contained in Regulatory Guide 1.16, "Reporting of Operating Information," Revision 4 - August 1975, and 5) the guidance contained in the NEI Guideline for Managing NRC Commitments, Revision 2, December 1995.

Technical Specifications 6.9.1.4 and 6.9.1.5 require that the following reports for the previous calendar year be submitted prior to March 1 of each year:

- A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/year and their associated man-rem exposure according to work and job functions.

- Documentation of all challenges to main steam line safety/relief valves.

- The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The limits are, "Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131," and "Less than or equal to 100/E-Bar microcuries per gram."

Technical Specification 4.8.1.1.3 requires reporting of all diesel generator failures, valid or non-valid. This report is made pursuant to Specification 6.9.1, "Routine Reports."

Regulation 10CFR50.59 requires that licensees submit, as specified in 10CFR50.4, a report containing a brief description of any changes, tests or experiments, including a summary of the safety evaluation of each. The report may be submitted annually or at shorter intervals.

Regulatory Guide 1.16 states that routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to March 1 of each year. Each annual operating report should include:

- A narrative summary of operating experience during the report period relating to the safe operation of the facility, including safety-related maintenance not covered elsewhere.

- For each outage or forced reduction in power of over 20 percent of design power level where the reduction extends for more than four hours:

  (a) The proximate cause and the system and major component involved (if the outage or forced reduction in power involved equipment malfunction).

(b) A brief discussion (or reference to reports) of any reportable occurrences pertaining to the outage or reduction.

(c) Corrective action taken to reduce the probability of recurrence, if appropriate.

(d) Operating time lost as a result of the outage or power reduction.

(e) A description of major safety-related corrective maintenance performed during the outage or power reduction, including system and component involved and identification of the critical path activity dictating the length of the outage or power reduction.

(f) A report of any single release of radioactivity or single exposure specifically associated with the outage which accounts for more than ten percent of the allowable annual values.

- A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions.

- Indications of failed fuel resulting from irradiated fuel examinations, including eddy current tests, ultrasonic tests, or visual examinations completed during the report period.

The NEI Guideline for Managing NRC Commitments is a commission-endorsed method for licensees to follow for managing or changing NRC commitments. As part of this process and for commitments that satisfy one of the five NEI decision criteria not involving a codified regulatory process, the NEI guidance specifies periodic staff notification, either annually or along with the FSAR updates as required by 10CFR50.71(e).

The NEI guideline further specifies that commitments dispositioned through the NEI process that satisfy none of the NEI decision criteria do not need to be reported in the licensee's periodic report because their regulatory and safety significance is negligible.

2.1 Annual Personnel Monitoring and Exposure Report

This section contains information pertaining to personnel radiation exposure and is included pursuant to Technical Specifications 6.9.1.5.a and Regulatory Guide 1.16, Section C.1.b.(3).

These values are estimated doses for the listed activities based on direct reading dosimeter data. No correction has been applied to these readings.

10 CFR PART 20

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Facility: 02

Report for Calendar Year: 1996

This report was produced with direct reading dosimeter data

Number of Persons Receiving Over 100 millirem is 797 Total MAN-REM: 349.890

		Number of Individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contractors and Others
OPERATIONS AND SURVEILLANCE	Maintenance Personnel	69.49	3.43	91.31	17.351	0.595	14.867
	Operating Personnel	31.86	3.22	0.83	15.795	0.381	0.163
	Health Physics Personnel	16.45	0.83	3.68	3.429	0.093	0.931
	Supervisory Personnel	10.23	1.05	4.04	1.299	0.087	1.065
	Engineering Personnel	13.84	8.38	25.24	2.806	1.189	6.632
ROUTINE MAINTENANCE	Maintenance Personnel	55.30	0.99	199.90	43.856	0.664	121.330
	Operating Personnel	1.43	0.00	0.17	4.080	0.000	0.414
	Health Physics Personnel	17.34	0.67	45.17	9.943	0.285	16.377
	Supervisory Personnel	2.65	0.38	5.44	3.569	0.046	2.599
	Engineering Personnel	5.82	6.32	17.18	2.969	2.392	6.566
INSERVICE INSPECTION	Maintenance Personnel	0.55	0.00	6.34	0.316	0.000	4.192
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.02	0.00	0.07	0.004	0.000	0.008
	Supervisory Personnel	0.02	0.00	0.07	0.005	0.000	0.079
	Engineering Personnel	0.10	0.33	0.72	0.031	0.149	0.527
SPECIAL MAINTENANCE	Maintenance Personnel						
	Operating Personnel						
	Health Physics Personnel			(See attached sheets)			
	Supervisory Personnel						
	Engineering Personnel						
WASTE PROCESSING	Maintenance Personnel	0.71	0.42	0.06	0.482	0.592	0.024
	Operating Personnel	0.00	0.00	0.00	0.025	0.000	0.000
	Health Physics Personnel	0.70	0.00	1.07	0.619	0.000	1.226
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.05	0.07	0.00	0.135	0.027	0.000
REFUELING	Maintenance Personnel	20.07	0.03	11.03	18.443	0.008	6.415
	Operating Personnel	0.40	0.00	0.00	0.837	0.000	0.000
	Health Physics Personnel	1.21	0.00	10.20	0.775	0.000	3.368
	Supervisory Personnel	3.59	0.86	0.07	1.351	0.228	0.003
	Engineering Personnel	0.75	2.94	3.52	0.241	0.590	0.910
TOTAL	Maintenance Personnel	159.74	4.87	353.05	84.496	1.859	160.022
	Operating Personnel	34.24	3.22	1.00	20.901	0.381	0.577
	Health Physics Personnel	39.58	1.50	60.19	15.918	0.378	21.910
	Supervisory Personnel	16.49	2.29	9.62	6.224	0.361	3.746
	Engineering Personnel	24.31	18.46	68.10	7.296	4.471	21.006
Grand Total		274.37	30.34	491.96	134.835	7.450	207.261

10 CFR PART 20

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Facility: 02

Report for Calendar Year: 1996

This report was produced with direct reading dosimeter data

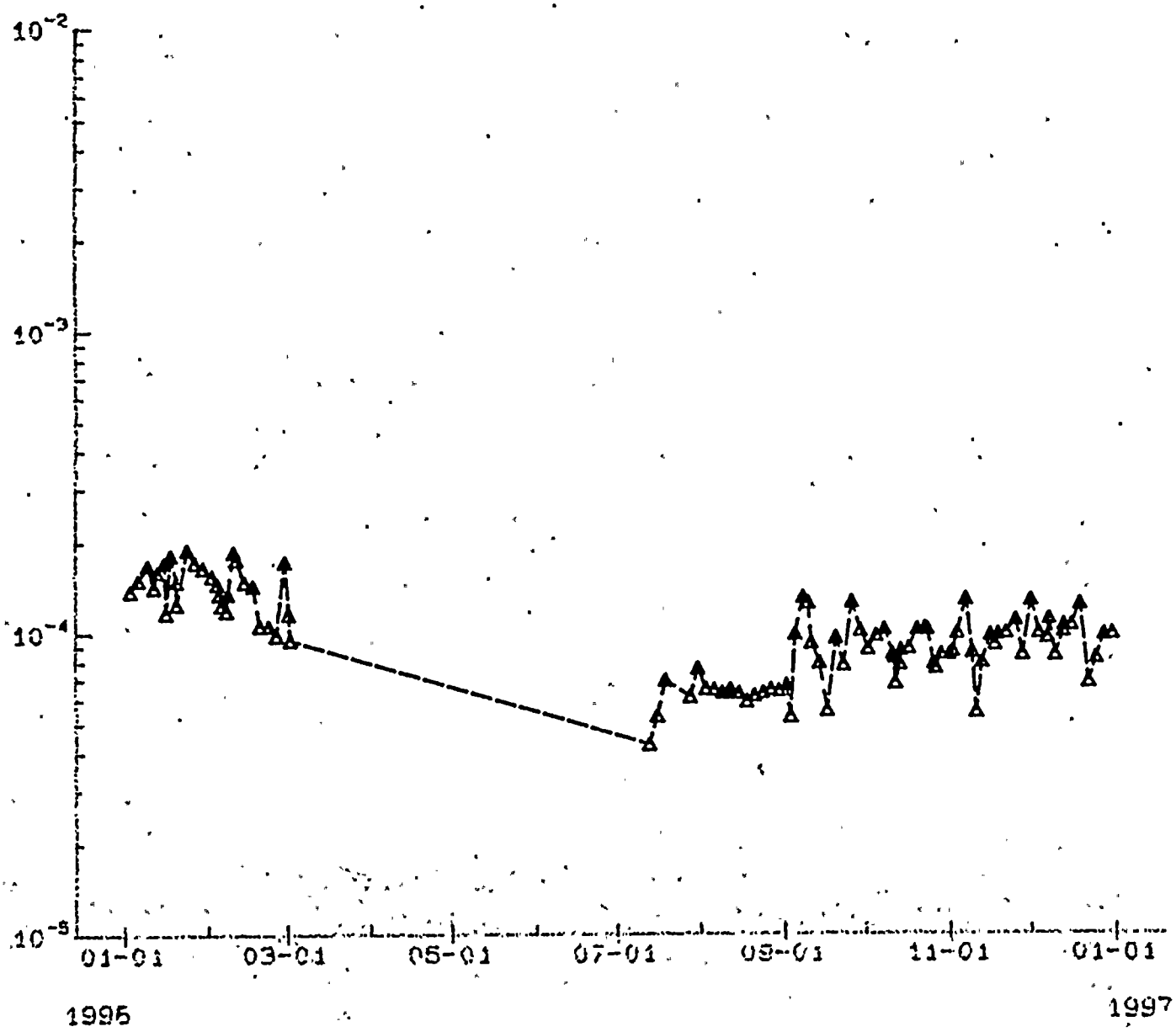
SPECIAL MAINTENANCE		Number of Individuals: Station Employees	Utility Employees	Contractors and Others	Year to Date Dose (MAN-REM): Station Employees	Utility Employees	Contractors and Others
1. Install Vibration Mitigation Clamp, Jet Pump Sensing Line	Maintenance Personnel	2.13	0.00	12.19	0.634	0.000	3.623
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	2.32	0.00	0.00	0.690	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.16	0.00	6.40	0.047	0.000	1.901
2. Reactor Recirculation and Residual Heat Removal Vibration Testing	Maintenance Personnel	0.00	0.00	12.39	0.000	0.000	3.681
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	6.52	0.000	0.000	1.937
3. Tube Plugging, Condenser/Heat. Exchanger 9, C Water Box	Maintenance Personnel	4.60	0.00	1.12	1.366	0.000	0.334
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.32	0.00	0.00	0.094	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	2.95	0.42	1.03	0.876	0.124	0.305
4. Adjustable Speed Drive Implementation	Maintenance Personnel	0.00	0.00	8.14	0.000	0.000	2.419
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
5. Source Range Monitor Drive B Gearbox Replacement	Maintenance Personnel	2.58	0.00	0.67	0.767	0.000	0.199
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.64	0.000	0.000	0.190
6. Adjust Main Steam Jet Pump Set Screw Gaps	Maintenance Personnel	0.00	0.00	1.54	0.000	0.000	0.457
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.40	0.00	0.00	0.118	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	1.06	0.000	0.000	0.315
7. Reactor Recirculation Valve 68A, Repair Leak	Maintenance Personnel	0.00	0.00	1.30	0.000	0.000	0.385
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.94	0.000	0.000	0.279
8. Remove Stellite Control Rod Blade Rollers/Velocity Limiters and Clean Spent Fuel Pool	Maintenance Personnel	0.00	0.00	0.75	0.000	0.000	0.222
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.48	0.00	0.00	0.142	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.71	0.000	0.000	0.210

9. ABB Fuel Debris Filter Removal	Maintenance Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Operating Personnel	0.55	0.00	0.00	0.164	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.30	0.00	0.00	0.090	0.000	0.000
10. Paint/Label Reactor Building RHR A&B, '422	Maintenance Personnel	1.29	0.00	0.00	0.383	0.000	0.000
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
11. '501 Reactor Bldg, Coating Walls and Doors	Maintenance Personnel	1.17	0.00	0.00	0.348	0.000	0.000
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
12. Rework Penetration Seals, QC 1&2	Maintenance Personnel	0.00	0.00	0.73	0.000	0.000	0.217
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	0.70	0.000	0.000	0.208
13. Temporary Shielding for Equipment Drain & Residual Heat Removal Pipes	Maintenance Personnel	0.61	0.00	2.05	0.182	0.000	0.609
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.00	0.00	1.33	0.000	0.000	0.396
14. Miscellaneous Projects	Maintenance Personnel	1.24	0.00	3.53	0.368	0.000	1.048
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.35	0.00	0.00	0.104	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Engineering Personnel	0.34	0.00	2.12	0.101	0.000	0.630

2.2 Reactor Coolant Specific Activity Levels

This section contains information pertaining to reactor coolant dose-equivalent iodine. The specific activity of the primary coolant was significantly less than 0.2 microcuries per gram dose-equivalent I-131 and 100/E-Bar microcuries per gram as required by Technical Specification 3.4.5.

This data is provided solely for informational purposes and ease of reference. Technical Specification 6.9.1.5.c only requires reporting when the results of specific activity analysis of primary coolant exceed the limits of Specification 3.4.5.



2.3 Main Steam Line Safety/Relief Valve Challenges

This section contains information pertaining to main steam line safety/relief valve challenges and is included pursuant to Technical Specification 6.9.1.5(b).

The main steam line safety/relief valve challenges (actuation events) are shown on the following tables. The data includes all in-situ tests. For ease of reference, the following descriptive codes are used for each actuation or failure to actuate:

- **Type of Actuation**

  A = Automatic
  B = Remote Manual
  C = Spring

- **Cause/Reason for Actuation**

  A = Overpressure
  B = ADS or other safety
  C = Test
  D = Inadvertent (Accidental/Spurious)
  E = Manual relief

- **Reactor Operating Condition Prior to Lift**

  A = Construction
  B = Preoperational, startup or power ascension tests in progress
  C = Routine startup
  D = Routine shutdown
  E = Steady state operations
  F = Load changes during routine operation
  G = Shutdown (hot or cold), except refueling
  H = Refueling

- **Failures and Reports**

  A = Failure of electrical or other components not considered part of the valve assembly - No SRVS failure report required
  B = Failure of any part of the valve - SRVS failure report will be filed
  C = No failures occurred - No SRVS failure report required
  D = LER Submitted - Report LER number in Item 316
  E = NPRDS will be submitted

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0140	63790-00-0139	63790-00-0049	63790-00-0134	63790-00-0050
Component ID (Location)	MS-RV-1B	MS-RV-1C	MS-RV-1A	MS-RV-1D	MS-RV-2B
Date of Actuation (Mo/Da/Yr)	3/2/96	3/2/96	3/2/96	3/2/96	3/2/96
Time of Day (24 Hour Clock)	1257	1257	1257	1257	1257
Type of Actuation (Code)	C	C	C	C	C
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	D	D	D	D	D
Rx Power Level Prior to Lift (% Rated Thermal)	15	15	15	15	15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	919	919	919	919	919
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	4 SEC.	6 SEC.	4 SEC.	4 SEC.	6 SEC.
Failures, Reports (Code)	C	C	C	C	A
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0122	63790-00-0054	63790-00-0138	63790-00-0053	63790-00-0124
Component ID (Location)	MS-RV-2C	MS-RV-2A	MS-RV-2D	MS-RV-3B	MS-RV-3C
Date of Actuation (Mo/Da/Yr)	3/2/96	3/2/96	3/2/96	3/2/96	3/2/96
Time of Day (24 Hour Clock)	1257	1257	1257	1257	1257
Type of Actuation (Code)	C	C	C	C	C
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	D	D	D	D	D
Rx Power Level Prior to Lift (% Rated Thermal)	15	15	15	15	15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	919	919	919	919	919
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	4 SEC.	4 SEC.	4 SEC.	4 SEC.	6 SEC.
Failures. Reports (Code)	C	C,	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0058	63790-00-0126	63790-00-0137	63790-00-0056	63790-00-0135
Component ID (Location)	MS-RV-3A	MS-RV-3D	MS-RV-4B	MS-RV-4C	MS-RV-4A
Date of Actuation (Mo/Da/Yr)	3/2/96	3/2/96	3/2/96	3/2/96	3/2/96
Time of Day (24 Hour Clock)	1257	1257	1257	1257	1257
Type of Actuation (Code)	C	C	C	C	C
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	D	D	D	D	D
Rx Power Level Prior to Lift (% Rated Thermal)	15	15	15	15	15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	919	919	919	919	919
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	6 SEC.	4 SEC.	4 SEC.	4 SEC.	4 SEC.
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0060	63790-00-0136	63790-00-0062		
Component ID (Location)	MS-RV-4D	MS-RV-5B	MS-RV-5C		
Date of Actuation (Mo/Da/Yr)	3/2/96	3/2/96	3/2/96		
Time of Day (24 Hour Clock)	1257	1257	1257		
Type of Actuation (Code)	C	C	C		
Cause/Reason for Actuation (Code)	C	C	C		
Rx Operating Condition Prior to Lift (Code)	D	D	D		
Rx Power Level Prior to Lift (% Rated Thermal)	15	15	15		
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A		
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer		
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN		
Rx Pressure Prior to Actuation (PSIG)	919	919	919		
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A		
Duration of This Actuation (Minutes, Seconds)	4 SEC.	4 SEC.	6 SEC.		
Failures, Reports (Code)	C	C	C		
LER Number (5 Digit Number)	N/A	N/A	N/A		
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0052	63790-00-0126	63790-00-0055	63790-00-0061	63790-00-0059
Component ID (Location)	MS-RV-3C	MS-RV-3D	MS-RV-4B	MS-RV-4D	MS-RV-5B
Date of Actuation (Mo/Da/Yr)	5/28/96	5/28/96	5/28/96	5/28/96	5/28/96
Time of Day (24 Hour Clock)	N/A	N/A	N/A	N/A	N/A
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	G	G	G	G	G
Rx Power Level Prior to Lift (% Rated Thermal)	0	0	0	0	0
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	0	0	0	0	0
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	5 min	5 min	5 min	5 min	5 min
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0045	63790-00-0057	63790-00-0051	63790-00-0048	63790-00-0135
Component ID (Location)	MS-RV-1C	MS-RV-4C	MS-RV-3B	MS-RV-1A	MS-RV-4A
Date of Actuation (Mo/Da/Yr)	5/28/96	5/28/96	5/28/96	6/2/96	6/2/96
Time of Day (24 Hour Clock)	N/A	N/A	N/A	N/A	N/A
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	G	G	G	G	G
Rx Power Level Prior to Lift (% Rated Thermal)	0	0	0	0	0
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number.Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	0	0	0	0	0
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	5 min	5 min	5 min	5 min	5 min
Failures, Reports (Code)	A	A	A	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0062	63790-00-0051	63790-00-0045	63790-00-0057	63790-00-0051
Component ID (Location)	MS-RV-5C	MS-RV-3B	MS-RV-1C	MS-RV-4C	MS-RV-3B
Date of Actuation (Mo/Da/Yr)	6/2/96	6/2/96	6/8/96	6/8/96	6/8/96
Time of Day (24 Hour Clock)	N/A	N/A	N/A	N/A	N/A
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	G	G	G	G	G
Rx Power Level Prior to Lift (% Rated Thermal)	0	0	0	0	0
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	0	0	0	0	0
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	5 min	5 min	5 min	5 min	5 min
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?	Yes	Yes	Yes	Yes	Yes

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0045	63790-00-0048	63790-00-0051	63790-00-0052	63790-00-0126
Component ID (Location)	MS-RV-1C	MS-RV-1A	MS-RV-3B	MS-RV-3C	MS-RV-3D
Date of Actuation (Mo/Da/Yr)	6/16/96	6/16/96	6/16/96	6/16/96	6/16/96
Time of Day (24 Hour Clock)	0343	0346	0348	0350	0352
Type of Actuation (Code)	C	C	C	C	C
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	3	3	3	3	3
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	920	922	921	921	919
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	4 sec	4 sec	4 sec	4 sec	4 sec
Failures, Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0055	63790-00-0061	63790-00-0059	63790-00-0057	63790-00-0135
Component ID (Location)	MS-RV-4B	MS-RV-4D	MS-RV-5B	MS-RV-4C	MS-RV-4A
Date of Actuation (Mo/Da/Yr)	6/16/96	6/16/96	6/16/96	6/16/96	6/20/96
Time of Day (24 Hour Clock)	0353	0357	0359	0355	1224
Type of Actuation (Code)	C	C	C	C	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	3	3	3	3	15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	920	921	921	920	950
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	4 sec	4 sec	4 sec	4 sec	1 min, 20 sec
Failures, Reports (Code)	C	C	C	A	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0055	63790-00-0057	63790-00-0126	63790-00-0059	63790-00-0062
Component ID (Location)	MS-RV-4B	MS-RV-4C	MS-RV-3D	MS-RV-5B	MS-RV-5C
Date of Actuation (Mo/Da/Yr)	6/20/96	6/20/96	6/20/96	6/20/96	6/20/96
Time of Day (24 Hour Clock)	1228	1230	1250	1251	1253
Type of Actuation (Code)	B	B	B	B	B
Cause/Reason for Actuation (Code)	C	C	C	C	C
Rx Operating Condition Prior to Lift (Code)	C	C	C	C	C
Rx Power Level Prior to Lift (% Rated Thermal)	15	15	15	15	15
Time Req'd for Tailpipe Temp to Return to Normal	N/A	N/A	N/A	N/A	N/A
Other Instrumentation Type (Code)	Process Computer	Process Computer	Process Computer	Process Computer	Process Computer
Other Instrumentation Number Reading and Units	OPEN	OPEN	OPEN	OPEN	OPEN
Rx Pressure Prior to Actuation (PSIG)	950	950	950	950	950
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A	N/A	N/A	N/A	N/A
Duration of This Actuation (Minutes, Seconds)	1 min, 7 sec	1 min, 16 sec	1 min, 12 sec	1 min	1 min, 5 sec
Failures. Reports (Code)	C	C	C	C	C
LER Number (5 Digit Number)	N/A	N/A	N/A	N/A	N/A
Comments Regarding This Actuation Attached?					

NOTE: Includes all In-Situ Tests

For Each Actuation or Failure to Actuate:	1	2	3	4	5
S/R Valve Serial Number	63790-00-0061				
Component ID (Location)	MS-RV-4D				
Date of Actuation (Mo/Da/Yr)	6/20/96				
Time of Day (24 Hour Clock)	1622				
Type of Actuation (Code)	B				
Cause/Reason for Actuation (Code)	C				
Rx Operating Condition Prior to Lift (Code)	C				
Rx Power Level Prior to Lift (% Rated Thermal)	15				
Time Req'd for Tailpipe Temp to Return to Normal	N/A				
Other Instrumentation Type (Code)	Process Computer				
Other Instrumentation Number Reading and Units	OPEN				
Rx Pressure Prior to Actuation (PSIG)	950				
IF AVAILABLE/IF APPLICABLE					
Reseat Pressure At Valve Closure (PSIG)	N/A				
Duration of This Actuation. (Minutes, Seconds)	1 min. 1 sec				
Failures, Reports (Code)	C				
LER Number (5 Digit Number)	N/A				
Comments Regarding This Actuation Attached?					

2.4 Summary of Plant Operations

This section contains a narrative summary of operating experience and is included pursuant to Regulatory Guide 1.16, Sections C.1.b.(1) and C.1.b.(2).

January 1996

- At the beginning of the month, the plant was operating at full power. From January 12, 1996 through part of January 16, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 60 percent. Following economic dispatch, the plant returned to full power operation on January 16, 1996.

- On January 19, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 60 percent.

- Following economic dispatch, the plant returned to full power operation on January 22, 1996 and operated at or near full power for the remainder of the month.

February 1996

- At the beginning of the month, the plant was operating at full power. On February 3, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 80 percent.

- On February 4, 1996 the plant was maneuvered into a planned downpower for Main Condenser maintenance. Following condenser work, the plant returned to full power operation.

- On February 10 and 16, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 75 and 55 percent respectively.

- Following economic dispatch, the plant was returned to full power and operated at or near full power until February 27, 1996 when it was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 80 percent. The plant returned to full power operation (100 percent) on February 28, 1996.

- On February 28, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 80 percent. The plant returned to full power operation (100 percent) on February 29, 1996.

- On February 29, 1996 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was reduced to about 60 percent.

March 1996

- At the beginning of the month the plant continued to operate at reduced power due to economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power generation was about 70 percent.

- On March 2, 1996 the Main Turbine Generator was removed from service and the plant was shutdown for economic dispatch (load following) at the request of the Bonneville Power Administration. The plant remained in a reserve shutdown condition until the beginning of the annual planned maintenance and refueling outage.

April 1996

- At the beginning of the month the plant continued to be in a reserve shutdown condition. On April 13, 1996 the plant entered the annual maintenance and refueling outage.

May 1996

- The plant was in the annual maintenance and refueling outage for the entire month.

June 1996

- The plant ended the annual maintenance and refueling outage on June 21, 1996.

- Following restart, the plant was manually scrammed on June 24, 1996 due to an unexpected plant response during testing of a recently-installed Digital Feedwater Level Control System. The reason for the unexpected plant response was due to a digital reactor feedwater controller error. A change was made to the controller software and plant startup resumed.

- At the end of the month the plant was operating at 25 percent power and continuing with startup testing following the maintenance and refueling outage.

July 1996

- At the beginning of the month the plant was ramped up to 68 percent power. Power was maintained at this level due to problems encountered during testing of the recently-installed Reactor Recirculation System Pump Adjustable Speed Drives and the Digital Feedwater Level Control System.

- At the end of the month the plant was operating at 68 percent power and continuing with startup testing following the maintenance and refueling outage.

August 1996

- At the beginning of the month, startup testing following the maintenance and refueling outage was still in progress with the focus on the recently-installed Reactor Recirculation System Pump Adjustable Speed Drives and the Digital Feedwater Level Control System. Power was maintained at 64 percent power.

- On August 10, 1996 the plant continued to operate as a severe transient coursed through the interconnected electrical transmission grids in the Western United States.

- During the evening of August 10, 1996, on-line Reactor Recirculation (RRC) System Pump RRC-P-1B "ran back" from approximately 50 Hz to 15 Hz during testing on part of the Adjustable Speed Drive System for the pump. Reactor power decreased from 64 percent to 48 percent. Reactor power was subsequently restored to 65 percent.

- On August 11, 1996 an unanticipated increase of approximately two percent power occurred due to an apparent failure of a portion of the control for both reactor recirculation pumps. The part of the computer control involved in the failure was removed from the control circuitry and local reactor recirculation pump control was established. Testing and troubleshooting efforts were suspended pending a thorough review of a Reactor Recirculation System Pump Adjustable Speed Drives and Digital Feedwater Level Control System test plan.

- On August 29, 1996 following development of a revised plan, testing was resumed and the power level was increased to support the effort.

- At the end of the month the plant was operating at 69 percent power and testing continued.

September 1996

- The plant entered the month at about 69 percent power as testing of the Reactor Recirculation System Pump Adjustable Speed Drives and the Digital Feedwater Level Control System continued.

- On September 5, 1996, following successful testing efforts, reactor power was raised to 100 percent.

- On September 13, 1996 power was reduced to approximately 52 percent for economic dispatch (load following) at the request of the Bonneville Power Administration.

- On September 15, 1996, during routine turbine valve testing, a turbine stop valve failed to open. On September 16, 1996 power was reduced to 25 percent to perform repairs to the Turbine Stop Valve Control System.

- Following successful repair efforts, the plant resumed 100 percent power operation on September 17, 1996.

- On September 20, 1996 power was reduced to 60 percent power to plug a leaking tube in the Main Condenser and to replace some scram solenoid pilot valves. Following repairs, the plant resumed full power operation on September 22, 1996 and operated at or near 100 percent power for the remainder of the month.

October 1996

- The plant entered the month at 100 percent power. On October 9, 1996 power was reduced to approximately 55 percent to repair an Adjustable Speed Drive channel that experienced a failed resistivity meter. On October 10, 1996 the plant resumed 100 percent power

- On October 11, 1996 power was reduced to approximately 75 percent power to perform scram time and turbine valve testing, and then to 60 percent for digital feedwater system testing. Following a firmware change on each of the reactor feedwater pumps and subsequent testing, the plant resumed 100 percent power on October 13, 1996.

- The plant operated at or near full power until October 23, 1996 when power was reduced to repair an Adjustable Speed Drive channel that experienced a failed resistivity meter. During power ascension following repairs, the same channel tripped on overspeed.

- At the end of the month the plant was operating at 92 percent power.

November 1996

- The plant entered the month at approximately 92 percent power due to a channel failure in the Adjustable Speed Drive System. On November 1, 1996 power was reduced to approximately 49 percent to perform repairs on Adjustable Speed Drive Channel 1A/2. Following successful repair efforts, plant power was increased to 100 percent on November 3, 1996.

- On November 9, 1996 a power reduction to 75 percent was commenced to perform monthly turbine valve testing. During this evolution, Main Steam (MS) System Turbine Intercept Valve MS-V-165C would not re-open during testing. Accordingly, power was reduced to 62 percent as prescribed by the Technical Specifications. On November 10, power was further reduced to 55 percent to repair the valve. The air solenoid and internal o-rings were replaced.

- Following successful repair efforts, plant power was increased to 100 percent on November 13, 1996.

- With the exception of a few short downpowers for routine maintenance, the plant operated at or near 100 percent power during the remainder of the month.

December 1996

- The plant operated at or near 100 percent power during the month.

2.5 Significant Corrective Maintenance Performed on Safety-Related Equipment

This section contains a description of major, safety-related corrective maintenance performed during outages or power reductions and is included pursuant to Regulatory Guide 1.16, Section C.1.b(2)(e). The following descriptions consist of summaries of information provided through the Nuclear Plant Reliability Data System (NPRDS). In addition to safety-related equipment, components considered to be essential for power generation are also included.

- **APRM-F/U-A**

  During the annual maintenance and refueling outage, an alarm was received on the Average Power Range Monitor (APRM) "A" flow unit. During investigation of the problem, the "C" flow unit was taken to bypass and the "A" unit alarm cleared and did not return until after the "C" flow unit was removed from the bypass mode.

  The cause of the problem was indeterminate. The test monitor switch was replaced and tested. No further problems were noted. (Failure Date: 06/18/96)

- **CAC-EHO-FCV/4A**

  During testing of a Containment Atmosphere Control (CAC) System flow control valve, it was discovered that a normally closed limit switch on the electro-hydraulic operator was in the open position. The cause of the problem was due to an incorrectly wired connection in the limit switch. The miswiring apparently occurred during previous maintenance activities.

  The wiring error was corrected and the switch was adjusted. No further problems were noted. (Failure Date: 02/24/96)

- **CAC-FCV-4B**

  During the performance of a local leak rate test, leakage in excess of allowable limits was discovered on Containment Atmosphere Control (CAC) System Valve CAC-FCV-4B. The cause of the problem was due to line and valve rust which degraded seat integrity.

  The rust was removed and the valve seat was machined and lapped. Following completion of a successful post-maintenance local leak rate test, no further problems were noted. (Failure Date: 05/16/96)

- **CAC-V-4**

During the performance of a local leak rate test, a leak rate in excess of allowable limits was discovered on Containment Atmosphere Control (CAC) System Valve CAC-V-4. The cause of the problem was due to line and valve rust which degraded seat integrity.

The rust was removed and the valve seat was machined and lapped. Following completion of a successful post-maintenance local leak rate test, no further problems were noted. (Failure Date: 05/18/96)

- **COND-DM-1B**

During power operation, Condensate (COND) System Demineralizer COND-DM-1B failed its resin bleed-through test. There was evidence of resin tracking across the rubber washers at the bayonet fitting on 35 of the septa.

New septa (35) were installed in the area immediately adjacent to the draft tube and no further problems were noted. (Failure Date: 08/13/96)

- **COND-DM-1E**

During routine inspections while at power operation, it was observed that the resin strainer differential pressure on Condensate (COND) System Demineralizer COND-DM-1E had increased from ten psid to 15 psid. Following troubleshooting efforts and additional inspections, it was discovered that the draft tube was loose and not latched. In addition, most of the septa on the inner ring were damaged. The cause of the problem was attributed to flow-induced vibration or other failure mechanism.

Welded locking devices were added to the draft tube and new septa were installed. No further problems were noted. (Failure Date: 02/27/96)

- **COND-HX-2A**

During full power operation, it was noted that the level control valve for Condensate (COND) System Heat Exchanger COND-HX-2A was open further than normal. This was an indication of a tube leak in the heat exchanger. The cause of the problem was attributed to normal wear.

The heat exchanger was drained and nondestructive examinations were performed. Leaking tubes were plugged and no further problems were noted. (Failure Date: 03/04/96)

- **COND-HX-9**

  During routine sampling at full power operation, an increase in reactor sulfate levels was observed. The reason for the increase was determined to be a condenser tube leak in Condensate (COND) System Heat Exchanger COND-HX-9. The exact cause of the tube leak was indeterminate.

  The tube was plugged and no further problems were noted. (Failure Date: 02/02/96)

- **COND-HX-9**

  During main condenser inspection efforts, damage was noted on spray baskets, welds and a strong back for Condensate (COND) System Heat Exchanger COND-HX-9. The cause of the problem was attributed to normal usage and stresses.

  All damaged components were either repaired or replaced and no further problems were noted. (Failure Date: 05/07/96)

- **COND-HX-9**

  During power operation, an increase in reactor sulfate levels were observed. The reason for the increase was determined to be a condenser tube leak in Condensate (COND) System Heat Exchanger COND-HX-9. The exact cause of the tube leak was indeterminate.

  The tube was plugged and no further problems were noted. (Failure Date: 09/10/96)

- **COND-P-2A**

  During power operation, inboard and outboard seal leakage was discovered on Condensate (COND) System Pump COND-P-2A. The cause of the problem was determined to be a casting defect in the bearing assembly that resulted in premature failure of the bearing.

  A new bearing, bearing body and bearing caps were installed. Following completion of post-maintenance testing, no further problems were noted. (Failure Date: 08/18/96)

- **COND-P-2B**

  During power operation, mechanical seal leakage was observed on Condensate (COND) System Pump COND-P-2B. The cause of the problem was determined to be normal usage and wear.

The inboard and outboard seals and the outboard shaft sleeve were replaced. No further problems were noted. (Failure Date: 07/31/96)

- **COND-PCV-40**

During the annual maintenance and refueling outage, it was noted that the valve stem was worn on Condensate (COND) System Pressure Control Valve COND-PCV-40. The cause of the problem was attributed to normal usage and wear.

The valve stem, plug, seat ring, gaskets and packing were replaced. Following completion of successful diagnostic testing, no further problems were noted. (Failure Date: 04/23/96)

- **COND-V-141A**

During the annual maintenance and refueling outage, Engineering personnel were informed by the vendor for Condensate (COND) System Valve COND-V-141A that the potential existed for failure of the anti-rotation collar due to key misalignment or lack of a set screw. The cause of the problem was attributed to manufacturer assembly practices.

A new key was fabricated and installed on the valve. (Failure Date: 05/21/96)

- **CRA-42-8B6B**

During power operation, an increased temperature was observed on a load side fuse clip in Containment Return Air (CRA) Motor Controller CRA-42-8B6B for a fan in the primary containment cooling system. The cause of the problem was that the fuse clip had separated from the bakelite support, causing a higher resistance which resulted in the temperature increase.

The fuse block was replaced and no further problems were noted. (Failure Date: 09/24/96)

- **CRA-FN-5B**

During the annual maintenance and refueling outage, Maintenance personnel observed Containment Return Air (CRA) Fan CRA-FN-5B rotating in the incorrect direction during performance of preventive maintenance. The cause of the problem was due to reversed phase lead connections. The lead reversal apparently occurred during previous maintenance activities.

The leads were re-connected to the correct configuration and no further problems were noted. (Failure Date: 05/16/96)

- **CRD-HCU-2215**

During the annual maintenance and refueling outage, Control Rod Drive (CRD) System Hydraulic Control Unit CRD-HCU-2215 level switch failed to actuate during surveillance testing. The cause of the problem was attributed to mechanical binding in the actuating mechanism.

A new level switch was installed and no further problems were noted. (Failure Date: 03/28/96)

- **CRD-HCU-2623**

During operation with the plant at 80 percent power, water leakage past the piston was noted on the accumulator for Control Rod Drive (CRD) System Hydraulic Control Unit CRD-HCU-2623. The cause of the problem was attributed to normal wear or aging of the piston sealing mechanism.

The piston sealing mechanism was replaced and no further problems were noted. (Failure Date: 02/04/96)

- **CRD-HCU-3443**

During the annual maintenance and refueling outage, hydrogen leakage was noted past the packing and around the stem on a valve associated with Control Rod Drive (CRD) System Hydraulic Control Unit CRD-HCU-3443. The cause of the problem was attributed to normal wear.

The valve was replaced with a new valve and no further problems were noted. (Failure Date: 04/18/96)

- **CRD-P-1B**

During power operation, minor leakage was observed on the positive seal supply line and casing drain plug for Control Rod Drive (CRD) System Pump CDR-P-1B. The cause of the problem was attributed to normal wear.

The line was disassembled and a new union joint was installed. No further problems were noted. (Failure Date: 10/30/96)

- **DLO-P-3A2**

  During full power operation, a broken coupling occurred on Diesel Generator Lube Oil (DLO) Pump DLO-P-3A2. The cause was attributed to high vibration due to inadequate pump mounting supports. The high vibration resulted in increased stresses which led to crack development.

  The coupling was replaced and pumps of similar design were inspected and additional couplings were replaced. A re-design of the support structure was not performed. This decision was based on a cost benefit analysis. Couplings are to be inspected and replaced when necessary. (Failure Date: 01/05/96)

- **DLO-P-3B2**

  During operation with the plant at 60 percent power, a broken coupling occurred on Diesel Generator Lube Oil (DLO) Pump DLO-P-3B2. The cause was attributed to inadequate design for the effects of resonance, cold spring and differential thermal expansion on coupling alignment.

  The coupling was replaced and flex hoses were installed to dampen vibrations. No further problems were noted. (Failure Date: 02/18/96)

- **FDR-V-3**

  During the annual maintenance and refueling outage, leakage was observed on Radioactive Floor Drain (FDR) System Valve FDR-V-3. The cause of the problem was determined to be normal usage and wear. A contributing cause was line rust which degraded the seat.

  The valve was cleaned and a new seat was installed. (Failure Date: 04/26/96)

- **FDR-V-3**

  During the annual maintenance and refueling outage, Radioactive Floor Drain (FDR) System Valve FDR-V-3 failed to close during local leak rate testing. The cause of the problem was determined to be binding in the stem seal area. The stem nut had apparently been over-tightened during previous maintenance work.

  The stem nut was loosened and no further valve stroking problems related to this event were noted. (Failure Date: 05/19/96)

- **FDR-V-3**

During power operation, Radioactive Floor Drain (FDR) System Valve FDR-V-3 exceeded the closing time limit during stroke time testing. The cause of the problem was determined to be foreign material and debris that had collected in the seat area. The source was loop seal piping debris that lodged into the seat from increased flow during a drain down of the undervessel drywell FDR sump.

The valve was disassembled and cleaned. No further problems were noted with this particular valve. (Failure Date: 07/06/96)

- **FDR-V-4**

During power operation, Radioactive Floor Drain (FDR) System Valve FDR-V-4 exceeded the closing time limit during stroke time testing. The cause of the problem was attributed to foreign material and debris that had collected in the seat area. The source was from loop seal piping debris that lodged into the seat from increased flow during a drain down of the undervessel drywell FDR sump. A contributing cause was loose set screws that allowed the coupling to shift out of alignment.

The valve was disassembled and cleaned. Existing set screws were tightened and an additional set were installed. No further problems were noted. (Failure Date: 08/01/96)

- **HPCS-P-3**

During power operation, high vibration readings were noted on the bearings for High Pressure Core Spray (HPCS) System Discharge Piping Fill Pump HPCS-P-3. The cause of the problem was determined to be normal wear leading to bearing degradation.

The pump was disassembled and the shaft, sleeve, bearings, seals and o-rings were replaced. No further problems were noted. (Failure Date: 09/11/96)

- **IRM-EMSQ-601F**

During testing efforts while at power operation, it was noted that positive and negative adjustments could not be made to Intermediate Range Monitor (IRM) 15-volt Power supply IRM-EMSQ-601F. The cause of the problem was attributed to a defective circuit card in the power supply.

The power supply was replaced and no further problems were noted. (Failure Date: 08/01/96)

- **LPRM-DET-40/57**

  During power operation, several spurious alarms were observed with Local Power Range Monitor (LPRM) LPRM-DET-40/75. The cause of the problem was attributed to a defective auxiliary circuit card.

  The auxiliary circuit card was replaced and no further problems were noted. (Failure Date: 02/04/96)

- **MS-AO-4C**

  During the annual maintenance and refueling outage, an air leak was observed between the valve and manifold for Main Steam (MS) System Valve Air Operator MS-AO-4C. The cause of the problem was due to failed o-rings due to normal wear and usage.

  Four o-rings were replaced and no further problems were noted. (Failure Date: 05/28/96)

- **MS-PS-47A**

  During power operation, it was discovered that the setpoint for Main Steam (MS) System Pressure Switch MS-PS-47A was out of tolerance in the conservative direction. The cause of the problem was that the pressure switch case had not been vented.

  The pressure switch case was vented and no further problems were noted. (Failure Date: 11/26/96)

- **MS-PS-47B**

  During power operation, it was discovered that the setpoint for Main Steam (MS) System Pressure Switch MS-PS-47B was out of tolerance in the conservative direction. The cause of the problem was that the pressure switch case had not been vented.

  The pressure switch case was vented and no further problems were noted. (Failure Date: 11/26/96)

- **MS-PS-47C**

  During power operation, it was discovered that the setpoint for Main Steam (MS) System Pressure Switch MS-PS-47C was out of tolerance in the conservative direction. The cause of the problem was that the pressure switch case had not been vented.

  The pressure switch case was vented and no further problems were noted. (Failure Date: 11/26/96)

- **MS-PS-47D**

  During power operation, it was discovered that the setpoint for Main Steam (MS) System Pressure Switch MS-PS-47D was out of tolerance in the conservative direction. The cause of the problem was that the pressure switch case had not been vented.

  The pressure switch case was vented and no further problems were noted. (Failure Date: 11/26/96)

- **MS-V-165C**

  During testing efforts while at power operation, it was noted that Main Steam (MS) System Intercept Valve MS-V-165C would not re-open when a test button was released. The cause of the problem was attributed to poor manufacturing practices resulting in particulates in the assembly. The particulates caused binding in the solenoid.

  The air solenoid valve was replaced. (Failure Date: 09/15/96)

- **MS-V-165C**

  During power operation, it was again noted that Main Steam (MS) System Intercept Valve MS-V-165C would not re-open during testing. The cause of the problem was attributed to either particulate contamination or problems with the sealing o-rings.

  The air solenoid and o-rings internal to the valve were replaced. No further problems were noted. (Failure Date: 11/09/96)

- **MS-V-37K**

  During the annual maintenance and refueling outage, it was noted that Main Steam (MS) System Valve MS-V-37K would not properly return to the full-closed position during vacuum breaker operability testing. The cause of the problem was attributed to foreign material buildup (hardened lubricant) at the hinge area which resulted in valve binding.

  The material buildup was filed down and no further problems were noted. (Failure Date: 05/03/96)

- **MS-V-37V**

  During the annual maintenance and refueling outage, it was noted that Main Steam (MS) System Valve MS-V-37V would not properly return to the full-closed position during vacuum breaker operability testing. The cause of the problem was attributed to foreign material buildup (hardened lubricant) at the hinge area which resulted in valve binding.

The material buildup was filed down and no further problems were noted. (Failure Date: 05/03/96)

- **RCIC-MO-110**

During the annual maintenance and refueling outage, the yoke-to-valve collar for Reactor Core Isolation Cooling (RCIC) System Valve Motor Operator RCIC-MO-110 was found to be loose and the operator could be rotated by hand. The cause of the problem was attributed to vibration.

The yoke was torqued to 110 ft-lbs and no further problems were noted. (Failure Date: 06/02/96)

- **RCIC-V-66**

During local leak rate testing in the annual maintenance and refueling outage, leakage in excess of allowable limits was discovered for Reactor Core Isolation Cooling (RCIC) System Valve RCIC-V-66. The cause of the problem was attributed to a loose packing follower and an eroded shaft and carbon bushing due to normal aging and abnormal wear during usage.

The shaft, bushing and packing set were replaced. No further problems were noted. (Failure Date: 04/16/96)

- **RFW-DT-1B**

During the annual maintenance and refueling outage, it was discovered that the inboard bearing for Reactor Feedwater (RFW) System Turbine RFW-DT-1B prematurely failed. The cause of the problem was indeterminate.

New bearing pads and housing were installed and no further problems were noted. (Failure Date: 04/27/96)

- **RFW-FR-607**

During power operation, it was noted that a pen would stick on Reactor Feedwater (RFW) System Flow Recorder RFW-FR-607 when either increasing or decreasing flow. The cause was isolated problems with the rotor assembly.

The pen slide bar was cleaned and the rotor assembly replaced. No further problems were noted. (Failure Date: 09/11/96)

- **RFW-LS-624B**

  During power operation, a Reactor Pressure Vessel Level-8 trip signal was received from Reactor Feedwater (RFW) System Level Switch RFW-LS-624B. The signal was received concurrent with a Station Battery B1-2 ground alarm. The cause of the problem was indeterminate.

  The level switch was replaced and no further problems were noted. (Failure Date: 07/08/96)

- **RHR-DPIS-12A**

  During surveillance testing while at power operation, it was noted that Residual Heat Removal (RHR) System Differential Pressure Switch RHR-DPIS-12A could not be properly calibrated. The cause of the problem was attributed to normal aging and usage.

  The pressure switch was replaced and no further problems were noted. (Failure Date: 08/22/96)

- **RHR-P-3**

  During power operation, Residual Heat Removal (RHR) System Water Leg Pump RHR-P-3 tripped on electrical thermal overload. The cause of the thermal overload trip was due to a failure of the pump thrust bearing. The vibration-induced fatigue failure of the bearing was due to inadequate design and service application.

  The pump bearings and shaft were replaced with components of an updated design. No further problems were noted. (Failure Date: 10/16/96)

- **RHR-TRS-601**

  During power operation, it was observed that the display was gradually failing on Residual Heat Removal (RHR) System Temperature Recorder RHR-TRS-601. The cause of the problem was attributed to normal aging and usage.

  A new display was installed and no further problems were noted. (Failure Date: 06/24/96)

- **RPS-EPA-3E**

  During surveillance testing while at Hot Standby, it was noted the Reactor Protection System (RPS) Electrical Protection Assembly RPS-EPA-3E could not be calibrated. The cause was isolated to a problem with the logic board.

The logic board was replaced and no further problems were noted. (Failure Date: 03/07/96)

- **RPS-RLY-K16B**

During the annual maintenance and refueling outage, it was noted that Reactor Protection System (RPS) Relay RPS-RLY-K16B was in a degraded condition and failing. The cause of the problem was attributed to normal aging and usage.

The relay was replaced and no further problems were noted. (Failure Date: 05/14/96)

- **RRC-PS-18A**

During testing while at power operation, it was noted that the as-found trip setpoint for Reactor Recirculation (RRC) System Pressure Switch RRC-PS-18A was well below the administrative limit and could not be restored to within the required tolerances. The cause of the problem was indeterminate.

The pressure switch was replaced and no further problems were noted. (Failure Date: 08/15/96)

- **SLC-LT-1**

During power operation, it was observed that Standby Liquid Control (SLC) Level Transmitter SLC-LT-1 appeared to be providing erroneous indication of SLC tank level. The cause of the problem was due to a plugged sensing line (bubbler tube) which resulted in air backpressure influencing the output of the level transmitter.

The bubbler tube was rodded out and the transmitter was returned to service. No further problems were noted. (Failure Date: 10/06/96)

- **SLC-TS-3**

During surveillance testing while at 58 percent power, it was noted that Standby Liquid Control (SLC) System Temperature Switch SLC-TS-3 could not be calibrated. The cause of the problem was indeterminate.

The temperature switch was replaced and no further problems were noted. (Failure Date: 02/04/96)

2.6 Fuel Performance

This section contains information relative to fuel integrity. This input is provided solely for informational purposes and ease of reference. There were no indications of failed fuel during 1996. Regulatory Guide 1.16, Section C.1.b.(4), only requires reporting where, based on examination, there are indications of failed fuel.

Background

During 1995 the Supply System modified a WNP-2 FSAR commitment pertaining to surveillance of post-irradiated fuel. As part of our routine fuel inspection program that was described in the WNP-2 FSAR, a visual examination was to be performed on five to ten percent of the highest burnup assemblies of the discharged fuel after each refueling. The visual examination was for the detection of indications of generic gross cladding defects or anomalies that may have occurred during operation. This commitment was accepted by the NRC in the WNP-2 Safety Evaluation Report, as adequately addressing the issue of post-irradiation surveillance.

As an alternate approach, the Supply System evaluated post-irradiation fuel inspection activities and determined that it would be acceptable to perform visual inspection only on discharged fuel where there was indication of either actual or suspected gross cladding defects or anomalies. Examples of such indications include increased Offgas System activity and negative impacts on water chemistry parameters. This change to the post-irradiation surveillance program was incorporated into Amendment 50 (August 1995) to the WNP-2 FSAR.

1996 Results

Based on plant operational indicators, there was no evidence of fuel performance problems during Cycle 11. Accordingly, a visual inspection of the discharged fuel was determined to be unnecessary.

2.7 10CFR50.59 CHANGES, TESTS AND EXPERIMENTS

This section contains summaries of the Safety Evaluations (SE) completed for activities implemented during 1996 and is included pursuant to 10CFR50.59.

Federal Regulation 10CFR50.59 and Supply System 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 approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question.

A proposed change, test or experiment is deemed to involve an unreviewed safety question if 1) the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased, or 2) a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or 3) the margin of safety as defined in the basis for any technical specification is reduced.

Each change summarized in the following sections was evaluated and determined to neither represent an unreviewed safety question nor require a change to the WNP-2 Technical Specifications.

In certain instances, a single safety evaluation was used for several implementing activities. This is allowed by procedure only where an existing evaluation adequately covers the specific change being considered. If the activity extends beyond the plant mode bounds, then a separate evaluation is required. A separate evaluation is also required if out-of-service equipment, equipment lineups, modifications or temporary alterations are in place that invalidate the existing evaluation.

2.7.1 Plant Modifications

The section contains information pertaining to implemented Plant Modification Records (PMRs) and is included pursuant to 10CFR50.59.

- **PMR 86-0525-7** (SE 95-084)

  This PMR provided for snubber optimization of Main Steam System, Loop C, inside the primary containment drywell and containment exhaust purge bypass piping in the reactor building. This modification involved the removal and subsequent replacement of snubbers with rigid struts.

  It was concluded from the safety evaluation that the affected piping systems, components and supports were reanalyzed and remain qualified to applicable ASME Code and WNP-2 design basis requirements. Pressure boundary integrity of the piping systems and primary containment are maintained and engineered safety features used in mitigating transients will remain operable.

  The only possible accidents are those associated with reactor pressure boundary and containment isolation. Since the piping and component analyses meet all design requirements and no new pipe breaks are created in main steam piping, then the pressure boundary is maintained. The containment exhaust purge piping is exempt from postulated pipe rupture due to the ASME exemption criteria of being less than or equal to one-inch in diameter. All equipment will remain functional to maintain containment isolation.

- **PMR 87-0244-6** (SE 93-200)

  This PMR provided modification of the Reactor Recirculation (RRC) System by installing Adjustable Speed Drives (ASD) to control recirculation pump speed. This modification also provided direction for ASD interconnections to the Main Control Room. The ASD System is a functional replacement for a hydraulically-controlled flow control valve arrangement.

  It was concluded from the safety evaluation that the RRC System has no active safety function. Its primary relation to the design analyses is as an initiator of events. This modification does not alter assumptions pertaining to RRC pump response during transients or accidents. The coast-down characteristics are similar to all previous analyses of recirculation system events.

The design of the ASD interlocks and limiters will prevent operation of the system outside of acceptable operating parameters. Installation activities would occur during the refueling outage. Any potential failures that could occur during post-modification testing are bounded by previous analyses.

- **PMR 87-0282-0** (SE 95-098)

This PMR provided for the installation of an additional steam tunnel fan/coil air cooling unit to provide for improved peak summer load performance and as backup protection for the existing units.

It was concluded from the safety evaluation that the seismically-mounted backup unit provides no safety function. The unit would be installed to meet applicable design requirements preserving the important to safety function of equipment located in the vicinity. All associated mechanical and electrical components would also be installed to appropriate standards commensurate with this activity to ensure that no important to safety functions are affected.

No previously evaluated transients or operator post-accident responses are adversely impacted by the change.

- **PMR 89-0299-6** (SE 95-100)

This PMR provided for the replacement of flow transmitters in the Containment Atmosphere Control (CAC) System with units of an improved design.

It was concluded from the safety evaluation that the five new replacement transmitters would be procured to Quality Class 1 requirements and installed such that the original design function will be maintained. The new transmitters do not change the function or configuration of the CAC System in response to a design basis accident.

All measurement ranges, indications and recording and alarm functions remain the same as prior to the change.

- **PMR 91-0438-0** (SE 94-209)

This PMR provided for the replacement of Reactor Feedwater (RFW) analog control system components with a digital control system upgrade. This change was made due to shortcomings with the existing system, which had significant impact on plant performance. The RFW Level Control System controls the discharge of the RFW pumps into the reactor pressure vessel to maintain level within predetermined limits.

It was concluded from the safety evaluation that the modification to the RFW Level Control System and associated Turbine Governor Control System would not increase the probability of a previously evaluated transient or impact the outage safety plan. The RFW System is a non-safety related system that is used to control reactor vessel water level during normal operation. The system is physically isolated when the core is flooded during a refueling outage.

The level trips and plausible accidents are bounded by the automatic response of the Reactor Protection System. Implementation of the new level control system reduces operator burden by providing direct and consolidated control of turbine speed and enabling earlier introduction of the feedwater pumps during startup valve/feedwater pump crossover.

The change does not result in the possibility of increased challenges to important to safety systems. The upset range level indicator or feedwater flow indicators mentioned in Regulatory Guide 1.97 are not changed by this modification.

- PMR 92-0085-0 (SE 96-002)

This PMR provided for modification of Containment Atmosphere Control (CAC) System push-button "valve test" switches CAC-RMS-PBA and CAC-RMS-PBB on the hydrogen recombiner control panel. The push-buttons were replaced with maintained-contact control keylock switches. This change prevents the bypassing of the pressure suppression function of the drywell downcomers in the event of single test switch failure or inadvertent depressing of the push-button. This modification also corrected an operator "work-around" by elimination of the need to install a jumper around the CAC "valve test" switch during valve testing, rather than depressing the push-button for extended periods.

It was concluded from the safety evaluation that the new switches would not introduce any new failure modes that could increase the consequences of previously evaluated transients. The keylock feature was added to prevent inadvertent actuation during normal plant operation.

There is no impact on design basis bypass leakage rates. The addition of two switches in series serves as an additional barrier to single failure of any one switch. The replacement device serves an equivalent function, with improved performance characteristics.

- **PMR 92-0209** (SE 95-102)

This PMR provided for installation of Jet Pump Sensing Line (JPSL) mitigation support assemblies on the sensing lines of each of the 20 existing jet pump diffusers. Four of the JPSL supports were required on each of the sensing lines. The purpose of this modification was to reduce the resonant vibration responses of the JPSLs by shifting the natural frequencies of the line into a band which will not coincide with the vane passing frequency associated with recirculation pump speed.

It was concluded from the safety evaluation that this modification improves the security of the JPSLs and reduces the probability of sensing line failure. The mitigation clamps have been seismically analyzed to ensure they would not fail in a manner that could damage safety-related structures in the event of an earthquake.

Based on the results of stress analysis, loose parts analysis, and material compatibility it was determined that there is no credible failure mode associated with the clamp assembly that could impact previously evaluated accidents.

- **PMR 93-0049-0** (SE 95-015)

This PMR provided for modification of the control circuitry for High Pressure Core Spray System (HPCS) Valves HPCS-V-10 and HPCS-V-11 by the addition of time-delay on the drop-out timers. This modification will allow trapped rotor flux to decay following movement to a full-open position and prevent inadvertent circuit breaker tripping.

It was concluded from the safety evaluation that the safety function of these valves would not be impacted by the change. The valves are normally closed, except during testing, and the passive component function (electrical integrity) is maintained with the new relays. There would be no change in HPCS accident response time or support of secondary containment bypass leak rate.

The added time delay simply prevents inadvertent circuit breaker tripping when the valves are immediately taken to the closed position when initially traveling in the open direction.

- **PMR 93-0065-O** (SE 96-101)

This PMR provided for deletion of the lead-lag function of Radwaste Building Chillers WCH-CR-51A and WCH-CR-51B and modification of the control circuitry to allow the two chillers to run independently. This change will allow the chillers to be controlled independently from their own individual chilled water sensors, rather than from a common sensor.

It was concluded from the safety evaluation that the availability of the chillers will not be affected by removal of the lead-lag function from the control system. The cooling function normally provided by the chillers would be assumed by the Service Water System until such time that the units were returned to service.

The affected equipment is non-safety related, does not have an accident mitigation function, does not have an off-site dose reduction function and is not a credited system which supports equipment important to safety.

- **PMR 93-0157-2** (SE 95-097)

This PMR provided for replacement of Reactor Building to Wetwell Vacuum Relief Valves CSP-V-5, CSP-V-6 and CSP-V-9. The valves were replaced with components of an improved design to more reliably meet allowable leakage requirements.

It was concluded from the safety evaluation that the safety function of these CSP valves is to prevent excessive vacuum from developing in primary containment from such cases as inadvertent containment spray actuation. The new valves were procured and installed to Quality Class 1 and Seismic Category 1 standards.

There was no change in valve opening time, leak-tightness, quality and seismic requirements. System interfaces were also unchanged. There were no changes from the original valve function, other than a seal design which will allow for a tighter seal. The ability to seal consistently when in a containment isolation mode would decrease the consequences of an accident.

- **PMR 94-0057-0** (SE 94-074)

This PMR provided for mechanically blocking Reactor Recirculation (RRC) System Flow Control Valves RRC-V-60A and RRC-V-60B in the full-open position and removal of the associated hydraulic system. This modification was performed in support of the installation of an Adjustable Speed Drive System on the RRC pump motors.

It was concluded from the safety evaluation that this change would not have an adverse impact on the reactor coolant pressure boundary. Stress analyses performed on the RCC piping show that piping stresses will not be increased as a result of modifying the flow control valves. The final configuration of the modification would not introduce any new failure modes or operational transients.

Reactor protection trip delay time and pump/motor inertia would also not change. The Adjustable Speed Drive System was designed to ensure that acceptable fuel thermal margins are maintained in the event of a reactor protection trip.

- **PMR 94-0332-1** (SE 95-103)

This PMR provided for the installation of a zinc injection system. The addition of depleted zinc reduces the buildup of Co-60 on primary piping and components. The modification consisted of the installation of a vendor-supplied skid for the injection of dissolved zinc oxide into the Reactor Feedwater (RFW) System.

It was concluded from the safety evaluation that zinc levels of less than or equal to 65 ppb would not have an adverse affect on the intergrannular stress corrosion cracking characteristics of primary system components. Based on information received from the vendor, it was determined that this modification would not impact RFW pump seals, RFW nozzles or RFW flow measurement instrumentation. There would also be no detrimental impact to the fuel assemblies.

Installation and testing of the skid would not interfere with plant operations. The skid is also isolatable from plant components by the use of double isolation valves.

- **PMR 94-0346-0** (SE 95-066)

This PMR provided for the rework and modification of fire-rated penetration seals to support current fire tests and restore the penetrations to an acceptable configuration. This effort was associated with an ongoing penetration seal upgrade project. The upgrade project was implemented based on the results of an analysis which revealed that not all of the installed configurations are supported by current fire tests or acceptable evaluations.

It was concluded from the safety evaluation that penetration seal design variations have no impact on design basis accidents, only events. The design of the seals is intended to mitigate the effects of fires and, in some installations, other design basis events. The changes made by this modification have no impact on the ability of safety systems to perform during accident conditions, nor do they increase challenges to safety systems.

The fire barriers and seals within the scope of this modification are currently declared inoperable and an hourly fire tour has been implemented as a compensatory action. The fire tours will not be removed until the seals and barriers have been declared operable.

During the restoration efforts, required safe shutdown systems and components will be maintained in accordance with licensing basis documents.

- **PMR 94-0364-O** (SE 96-006)

This PMR provided for installation of a permanent curtain shielding support structure around a portion of Reactor Recirculation (RRC) System, Loop "A" piping for ALARA considerations. The permanent structure eliminated the need to assemble tube-lock scaffolding and install shielding during every outage.

It was concluded from the safety evaluation that installation of the shielding and the shielding support system would not affect any system, structure or component that mitigates the consequences of a design basis accident. The shielding support structure meets Seismic Category 1M requirements to prevent adverse II/I interactions with important to safety systems, structures and components.

The shielding and support structure was also designed to withstand all applicable loads, including pipe breaks and missiles. The shielding, which is located in primary containment, does not restrict access to vital areas or otherwise impede actions to mitigate the consequences of design basis accidents.

- **PMR 94-0364-1** (SE 96-010)

This PMR provided for installation of a permanent curtain shielding support structure around a portion of Reactor Water Cleanup (RWCU) piping for ALARA considerations. The permanent structure eliminated the need to assemble tube-lock scaffolding and install shielding during every outage.

It was concluded from the safety evaluation that installation of the shielding and the shielding support system would not affect any system, structure or component that mitigates the consequences of a design basis accident. The shielding support structure meets Seismic Category 1M requirements to prevent adverse II/I interactions with important to safety systems, structures and components.

The shielding and support structure was also designed to withstand all applicable loads, including pipe breaks and missiles. The shielding, which is located in primary containment, does not restrict access to vital areas or otherwise impede actions to mitigate the consequences of design basis accidents.

- **PMR 95-0174-0** (SE 95-099)

This PMR provided for the installation of an iron injection system. The addition of iron oxalate reduces the buildup of Co-60 on primary piping and components. The modification consisted of a permanent stainless steel injection point that was welded to the 36-inch condensate booster pump suction line.

It was concluded from the safety evaluation that, although iron concentration will be increased in the reactor feedwater, the level would still be well below the design basis value of 5.0 ppb. Increasing the iron concentration from 0.1 ppb to 0.5 ppb would not adversely impact any safety-related structure, system or component.

It was determined that this modification would not impact the performance of reactor feedwater heaters or flow nozzles. The new piping that is installed will be compatible with the existing piping design pressures and temperatures. The increase in iron concentration will also not affect the integrity of the fuel cladding. The cladding would not become embrittled or experience any wall thinning.

- **PMR 95-0236-0** (SE 95-092)

This PMR provided for the replacement of eight top-entry Local Power Range Monitor (LPRM) detector assemblies with bottom-entry models of a better design. This modification was made to provide for improved efficiencies during refueling outages.

It was concluded from the safety evaluation that the new LPRM detectors meet or exceed the design requirements of the original detectors. All of the replacement components, including cable and connectors, are qualified for the environment in which they would be installed.

No activities are included which would impact systems and result in challenges to safety-related equipment. Following changes to the processing software to compensate for the different sensitivity of the new LPRM detectors, the function will be identical to the existing system.

- **PMR 96-0043-0** (SE 96-015)

This PMR provided for the modification to Radwaste Building Release Duct Radiation Monitor Sample Racks WEA-SR-25 and WEA-SR-25A to prevent overheating of the system blower. This change was made to increase the reliability of Sample Rack WEA-SR-25 by reducing flow restrictions to lower exhaust air sample blower operating temperature. The blower had been operating at a higher than design temperature and required periodic replacement due to premature failure. Loss of the blower would result in the sample racks becoming inoperable.

It was concluded from the safety evaluation that this modification would not alter safety-related system response to an accident condition. The change will ensure that the radiation monitors perform any required post-accident function for the duration of the transient. Environmental conditions within the rack area are bounded by current radiation and temperature limits. Post modification testing would also verify final operability of the sample racks.

During the implementation phase, compensatory alternate sampling methods would be implemented and the sample racks not returned to operable status until the modification and associated follow-up testing were completed.

- **PMR 96-0057-0** (SE 96-034)

This PMR provided for removal of the Demineralized Water (DW) System flush capability for the Residual Heat Removal (RHR) Loops "A" and "B" sample lines. This modification also provided for installation of additional isolation valves in the jet pump sample lines, between the DW System and Post Accident Sampling System (PASS). The reason for the modification was to minimize recurrence of contamination of the DW System through the PASS by means of the demineralized water flushing lines.

It was concluded from the safety evaluation that removal of a portion of the DW System flush capability slightly increases the potential for increased dose. However, calculations have shown this slight change in dose would have a negligible effect in total area dose. The potential increase in dose within the secondary containment envelope was reviewed and found to be within current design requirements and limits.

All credited accident mitigation equipment and systems would remain functional and operable with the implementation of this modification. The PASS function will not be impacted for post-accident sampling and the ability to obtain data for post-accident evaluation, sheltering or recovery actions would still be maintained.

2.7.2 Temporary Modifications and Instrument Setpoint Changes

This section contains information pertaining to implemented Temporary Modification Requests (TMRs) and Instrument Setpoint Change Requests (ISCRs) and is included pursuant to 10CFR50.59.

- **ISCR 1280** (SE 96-014)

  This ISCR provided for a change to the setpoints for the isolation signal due to condenser vacuum for Main Steam (MS) System Pressure Switches MS-PS-56A, MS-PS-56B, MS-PS-56C and MS-PS-56D. The new setpoints were changed by means of a calculation and will allow less loss of vacuum prior to initiating an MSIV isolation.

  It was concluded from the safety evaluation that allowing for less loss of vacuum prior to initiating an MSIV isolation would result in a reduction in operating margin. The low vacuum function is provided to isolate the main steam system in the event of a loss of main condenser vacuum which would remove the effective capability of the condenser as a heat sink.

  The reduction in operating margin would not increase the probability of a loss of condenser vacuum event. However, it could slightly increase the probability of an MSIV isolation event. It was determined that any contribution to the increase in the probability of the MSIV isolation event due to the decrease in operating margin caused by the more conservative setpoint is significantly less than the probability of the MSIV isolation event. In addition, this small increase in the probability of the MSIV isolation event does not contribute to an increase in frequency.

- **ISCR 1284** (SE 96-041)

  This ISCR provided for revision to the ANALYZE computer program configuration file to turn off the zero stabilizer function for stack monitor intermediate and high range detectors PRM-RE-1B and PRM-RE-1C. It was determined that the zero stabilizer function could fail at high count rates and result in an incorrect indication of system failure in the main control room.

  It was concluded from the safety evaluation that this system is a post-accident system with no control functions. Information from this system is used for decisions pertaining to accident follow-up actions. However, the information used by Operations personnel in these scenarios will not be affected by this change.

The change affects only the zero stabilization function of the multi-channel analyzer system and has no impact on any other part of the system or the plant. The zero stabilizer function of the stack monitor multi-channel analyzer provides automatic correction for small changes in the zero value of the spectrum. Turning off the function will simply disable the testing of the zero stabilizer range. It will not affect the gain stabilizer function. The gain stabilizer function provides all of the automatic control that is required to maintain the multi-channel analyzer within required tolerances.

This change has no affect on the gross count rate response of the system. Gross count rate indication is the information that is used for accident follow-up decisions.

- **TMR 95-105** (SE 95-105)

This TMR provided for modification of the suction and discharge lines for Radwaste Building HVAC (WOA) Sample Racks WOA-SR-18A, WOA-SR-18B, WOA-SR-19A and WOA-SR-19B to increase flow through the system. Increased flow was obtained by merging the intake and exhaust lines to the pump suction, and exhausting to the local atmosphere. The increased flow was necessary to reduce blower operating temperatures to preclude premature failure.

It was concluded from the safety evaluation that the modification will ensure that the blowers are operating within vendor recommendations to ensure availability during the required post-accident operating time. The control room remote air intake radiation monitors are not the initiator of any previously evaluated transient.

Based on testing results, it was concluded that system flow would be increased by approximately 1.5 times the current value, which dropped temperatures within the recommended operating range. With this modification, the blowers could be expected to operate reliably during post-accident conditions.

- **TMR 96-013** (SE 96-033)

This TMR provided for removal of a tab from the indicator shaft for Reactor Core Isolation Cooling (RCIC) System Valve RCIC-V-66 to improve the operating characteristics of the valve. With the tab removed, the indicator shaft will not turn when the hanger arm rotates on the actuator shaft. This would then allow for the ability to sufficiently tighten indicator shaft packing to prevent leakage.

It was concluded from the safety evaluation that the hanger arm and disc assembly of the valve would continue to function as designed to ensure rapid isolation in the event of an RCIC System line break. The hanger arm and disc assembly will move more freely, being independent of the indicator shaft.

Removal of the tab from the indicator shaft would not affect the system injection or containment isolation function of the valve. Because the indicator shaft will no longer turn in its packing, there is less probability of packing binding on the shaft or packing degradation due to wear.

- **TMR 96-023** (SE 96-066)

This TMR provided for installation of a filtration system near the Standby Service Water (SSW) spray ponds to remove suspended solids from the spray pond water. The side stream filtration system is designed to remove organic material and silt present in the water, which will assist in maintaining the required heat transfer capability of the SSW System heat exchangers and reduce water treatment costs.

It was concluded from the safety evaluation that the change is non-safety related. Potential impacts on important to safety systems and functions were evaluated and it was concluded none of those impacts would result in the inability of those systems and components to meet their safety function. The impacts evaluated relate to seismic effects, spray pond water inventor requirements, tornado effects and the effects of localized flooding.

The SSW system and ultimate heat sink would still be available with full capability to mitigate accidents.

- **TMR 96-029** (SE 96-100)

This TMR provided for increasing the voltage applied to a Traversing Incore Probe (TIP) indexer. Indexer "B" would not move beyond channel one. This modification would increase the applied torque to the indexer mechanism and restore proper operation.

It was concluded from the safety evaluation that the increase in motor voltage would not impact any of the analyzed transients. The modification involves increasing the voltage to the indexer motor to a maximum of 200 VAC for no more than five seconds.

The voltages and currents to be applied to the non-safety related indexer motor are within the design characteristics of the penetration module.

2.7.3 FSAR Changes

This section contains information pertaining to FSAR Licensing Document Change Notices (LDCNS) and FSAR Change Notices (SCNs) and is included pursuant to 10CFR50.59.

- **LDCN-FSAR-96-063** (SE 96-078)

  This LDCN provided for a change to the FSAR to allow for the use of optimum flow rate for iodine sampling and substitution of a local alarming continuous air monitor for a mobile monitoring system.

  It was concluded from the safety evaluation that dose avoidance during the course of an accident involving radiation releases would be enhanced by the change due to earlier and more reliable indications of iodine in the air. A more reliable continuous air monitor is also being substituted for the currently-installed unit.

  The use of an optimum flow rate for iodine sampling and implementation of an improved air monitoring system do not increase the probability or consequences of an accident previously evaluated.

- **LDCN-FSAR-96-068** (SE 96-091)

  This LDCN provided for a change to the FSAR to lower the minimum diesel generator engine bay temperature to 40 degrees fahrenheit. In addition, the description of the HVAC system for the High Pressure Core Spray (HPCS) System batteries was modified to allow for use of supplemental heaters or other means to maintain battery temperature greater than or equal to 60 degrees fahrenheit.

  It was concluded from the safety evaluation that lowering the diesel engine room temperature from 70 degrees to 40 degrees would not have an adverse affect on the starting capability or reliability of the engines. The diesel engine supplier provided information to the effect that, as long a the coolant and lube oil temperatures are maintained at or above 85 degrees by a "keep warm" system, the engines can achieve a ten-second start in an ambient temperature of 40 degrees.

  The battery manufacturer supplied information to the effect that the HPCS batteries will provide full-rated capacity as long as the electrolyte temperature is at or above 60 degrees.

  Because lowering of the room temperature has no adverse impact and the batteries will perform as required at 60 degrees, the components will continue to function as required in support of emergency core cooling system accident mitigation.

- **LDCN-FSAR-96-077** (SE 96-090)

  This LDCN provided for a change to the FSAR to reflect planned organizational changes within the Plant Support Services Department and the addition of new ALARA review criteria.

  It was concluded from the safety evaluation that, from an organizational perspective, the change only involves realignment of responsibilities. Current functions such as solid waste processing have not changed. Clear criteria for determining the need for an ALARA review for procedures was also provided.

  These changes are not in conflict with any licensing basis documentation or commitments. Although the review of procedures for ALARA considerations is an element of the ALARA Program, the specific review criteria are not part of any regulatory basis for radiological protection requirements.

- **LDCN-FSAR-96-079** (SE 96-097)

  This LDCN provided for a change to the FSAR to reflect updated actions that are to be taken following an earthquake.

  It was concluded from the safety evaluation that the changes are consistent or conservative with EPRI guidance and a draft NRC regulatory guide pertaining to earthquake planning and follow-up actions. The recommended actions would ensure that conservative shutdown decisions are made and that the reliability of structures, systems, and components following an earthquake is not reduced.

  These changes affect only the criteria used to initiate a controlled manual reactor shutdown. No hardware is impacted.

- **LDCN-FSAR-96-080** (SE 96-092)

  This LDCN provided for several changes to the Fire Protection Program to reflect updated compensatory measures and editorial enhancements.

  It was concluded from the safety evaluation that previous fire protection analyses had concluded that compensatory measures could be altered under certain circumstances. This change allows Control Room Operators to act as a fire tour for inoperable wet-pipe sprinkler system and Halon confinement barriers.

  The change also allows for eight hours to establish the operability of video and portable detection systems when sprinkler or detection systems are inoperable in high radiation or contaminated areas.

These changes are consistent with previously-approved safety evaluations and provide reasonable assurance that adequate compensatory measures will still be implemented for the observation of fire or fire hazards in the areas of inoperable fire protection equipment.

- **LDCN-FSAR-96-081** (SE 96-099)

This LDCN provided for a change to the FSAR to delete the requirement for Fast Flux Test Facility (FFTF) personnel on the Hanford Reservation to provide direct WNP-2 Control Room notification of a sodium oxide release. Procedural arrangements are in place between FFTF and Supply System personnel for timely notification of the WNP-2 Control Room in the event of a sodium oxide release.

It was concluded from the safety evaluation that changing the method of notifying the control room would have no impact on the probability or consequences of a previously-evaluated accident. The original assumption of 55 minutes to notify the WNP-2 Control Room is still valid and provides adequate time for personnel to either isolate the control room or put on portable breathing equipment.

By continuing to provide ample warning time, Supply System Operators would be able to respond accordingly and place the plant in a normal shutdown condition in the event of an accident.

- **LDCN-FSAR-96-086** (SE 96-096)

This LDCN provided for a change to the FSAR to reflect the downgrade of Standby Service Water (SSW) System Pumphouse Air Intake Fan POA-FN-2A from Quality Class-1 to Quality Class-Augmented (QC-A).

It was concluded from the safety evaluation that the fan provides no active support for any equipment important to safety. The safety-related cooling function for the standby service water pumphouse HVAC system does not require the operation of POA-FN-2A. The SSW pump and related equipment are cooled by a safety-related fan-coil unit.

The potential loss of the intake fan would not result in pumphouse ambient temperatures reaching the maximum normal operating limits for safety-related equipment. It was determined that a quality classification of QC-A is adequate for this component.

- **SCN 95-058** (SE 96-035)

This SCN provided for a change to the FSAR to reflect a revision to the Reactor Core Isolation Cooling (RCIC) System isolation time delay logic. Closure of RCIC System Primary Containment Isolation Valves RCIC-V-8, RCIC-V-63 and RCIC-V-76 is delayed by logic time delay relays to prevent inadvertent isolation from high flow during system

initiation when the steam flow to the turbine is momentarily above the high flow setpoint. The calculated time delay relay setpoint in each division was changed from a two-second nominal value to a three-second allowable value. Process and instrument loop accuracies that were previously not accounted for were included in the revised calculation.

It was concluded from the safety evaluation that this change would not impact the overall function and accident response of the RCIC System. The safety function of these relays during design basis event mitigation is to 1) isolate the system to limit mass energy blowdown for an RCIC System high energy line break, and 2) limit long-term secondary containment bypass leakage after a system trip following a loss of coolant accident.

The allowable value of three seconds is derived from, and bounded by, the upper analytical limit of four seconds used in the event blowdown calculation. The accident analysis assumes the four second delay (including process and loop inaccuracies) prior to the RCIC valves receiving a closure signal.

- **SCN 95-062** (SE 95-095)

This SCN provided for a change to the FSAR to describe changes to the new fuel storage vault. Deck plates were added to the vault racks and only allow fuel to be placed in alternate locations. These temporary plates constitute a template which limit placements of new fuel to alternate locations in the vault. This change allows for the safe and efficient handling of ABB re-load fuel.

It was concluded from the safety evaluation that this activity has the effect of markedly increasing the space between the adjacent fuel assemblies. The additional spacing renders a criticality accident considerably less than the low probability arrangement previously employed in these racks.

The proposed physical changes to the refuel floor to accommodate the change do not impact or increase the consequences of previously evaluated transients. Items addressed included heavy crane load paths and seismic requirements.

- **SCN 95-064** (SE 95-091)

This SCN provided for a change to the Fire Protection Program to reflect a revision of fire door surveillance requirements. The primary technical change consists of the performance of weekly position inspections for unlocked, unsupervised fire doors instead of the current inspection by routine operator tours.

It was concluded from the safety evaluation that the less frequent surveillance to nonessential and non-plant block doors will continue to ensure their operability. Nonessential fire doors, by definition, are not credited with ensuring safe fire shutdown.

The changes in surveillance frequency do not affect the operation of fire protection system equipment beyond that previously evaluated. Since the change does not alter fire door operation or create new failure modes, extending the period between surveillance testing would not lower the ability of the equipment to mitigate or prevent the propagation of fires.

- SCN 95-070 (SE 96-007)

This SCN provided for a change to the FSAR to reflect that the High Pressure Core Spray (HPCS) System diesel generator motor-driven air compressor is powered from Motor Control Center (MCC) MC-6B instead of the HPCS bus.

It was concluded from the safety evaluation that supply power from non-safety related MC-6B to electric motor driven Diesel Starting Air (DSA) System Air Compressor DSA-M-C/1C would not affect the consequences or probability of a previously evaluated accident. The air compressor is of an augmented quality class due to seismic requirements, but has no specific safety function.

Loss of the air compressor due to loss of MC-6B would not affect operation of the HPCS diesel generator. Safety-related Diesel Starting Air Receivers DSA-AR-1C and DSA-AR-2C are capable of maintaining system air pressure and capacity for starting of the diesel generator, regardless of the status of the air compressor.

- SCN 95-072 (SE 95-101)

This SCN provided for a change to the FSAR to reflect updated secondary containment bypass leakage paths. Some existing bypass paths were eliminated from consideration and others were added based on a technical evaluation. The overall allowed bypass leak rate of 0.74 scfh was not changed by this SCN.

It was concluded from the safety evaluation that revising the FSAR to reflect the potential bypass leakage paths for secondary containment would not impact previously evaluated transients. The overall allowed leakage was not changed. Since the leak rate was not modified, the offsite and control room dose consequences would not be affected.

The primary containment penetrations that were eliminated as secondary containment bypass leak paths were all analyzed to ensure that any leakage would be processed by the Standby Gas Treatment System. The valves eliminated from consideration as potential bypass leakage paths were containment isolation valves. The measured leakage would still be included in the Leak Rate Testing Program (Type B and Type C).

- SCN 96-005 (SE 96-063)

This SCN provided for a change to the FSAR and Emergency Plan to reflect current bases and methods for the controlling and monitoring of contamination.

It was concluded from the safety evaluation that the change would not involve systems, structures or components and does not impact the physical barriers that protect against the uncontrolled release of radioactivity. Controls have been established to ensure there is no detectable fixed or loose contamination outside of the Radiologically Controlled Area under normal and emergency conditions. The intent to control licensed material has not been modified.

The current mix of isotopes in dry active waste are such that the other detector types will provide greater assurance that licensed radioactive material would be controlled in a manner consistent with ALARA considerations.

- SCN 96-008 (SE 95-106)

This SCN provided for a change to FSAR drawings to reflect an update of fire area boundaries and fire barriers. In some cases, essential and nonessential fire barriers have been downgraded. However, the derated barriers are simply corrections or previous errors where no fire rating was actually required.

It was concluded from the safety evaluation that the proposed changes do not lower the ability of safety systems to perform during accident conditions. The changes in fire area boundaries do not affect the operation of, or become hazards to, accident mitigation equipment beyond that previously evaluated.

The proposed barrier changes do not increase radiological releases or exposures to control room personnel since configuration controls exist to ensure secondary containment and control room ventilation boundary penetration seals are present, independent of the fire rating of the barrier.

- SCN 96-009

(SE 96-030)

This SCN provided for changes to the FSAR and Emergency Plan to allow for removal of on-site Thermoluminescent Dosimeter (TLD) readers and to transfer the processing of exposure information to a certified local vendor.

It was concluded from the safety evaluation that dose avoidance during the course of an accident involving radiation releases would be based on direct readout dosimeters. The TLDs are processed as a follow-up after any dose has been received and the information is used as comparison to the direct dosimeter information.

The results of TLD measurement of external doses to workers or the public would be available from the vendor within 48 hours upon request of the information.

(SE 96-044)

This SCN also provided for a change to the FSAR to allow for deletion of reference to automatic radwaste drum processing (filling, storage and monitoring). The current method for drum processing consists of a manual operation.

It was concluded from the safety evaluation that radwaste drum processing is a manual activity that would not increase the probability or consequences of an accident. The processing activity does not affect any safety-related or important to safety plant equipment.

It was also concluded that the dose to the public from a manually filled drum would be the same as from the same drum filled in a remote manner.

- SCN 96-012 (SE 96-022)

This SCN provided for a change to the WNP-2 Physical Security Plan to allow security officers who have left the protected area to re-enter the protected area without being subjected to a metal detector search. In addition, the plan was revised to allow for the removal of protected and vital area keys from the protected area when they are under the control and custody of on-duty security force personnel.

Changing the security plan to allow on-duty armed security officers to re-enter the protected area without being subjected to metal detector searches was endorsed by the NRC in Generic Letter 96-02, "Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat." The staff considered that this change could be made to the security plans in accordance with the provisions of 10CFR50.54(p).

With regard to removal of keys from the protected area, it was concluded from the safety evaluation that the ability to protect vital equipment target sets continues to be maintained by prompt response to any vital area door alarm when an unauthorized access attempt is detected by the security system.

The key control requirements identified in 10CFR73.55(d)(9) are still being met and have been incorporated into the security plan. There are no 10CFR73.55(d)(9) stipulations that these keys cannot leave the protected area when under the control of a security officer. It was also concluded that the likelihood of an attempt at radiological sabotage is not a function of whether or not a vital area key is taken off-site while under the control of an authorized on-duty security force officer.

- **SCN 96-014** (SE 96-013)

This SCN provided for clarification and consistency pertaining to seismic qualification of equipment and components. The method for assessing valve clearance was also modified.

It was concluded from the safety evaluation that the consequences of a previously analyzed seismic or dynamic accident would not be increased by these changes. Consequences can only increase if the equipment or components analyzed are damaged by the event. The proposed changes only involve analytical methods for seismic and dynamic qualification of equipment and components. The use of acceptably conservative analytical methods precludes any reduction in safety margin for systems and components. No changes to hardware were made by this change.

The equivalent static method invoked as a replacement for the more rigorous, but less conservative dynamic method, would not result in a less conservative design. The proposed change invokes static factors that are conservative. The change to the use of good practice to establish margin for valve clearance assessment, instead of a nominal 25 percent margin, has no impact on system or component operability.

- **SCN 96-015** (SE 95-089)

This SCN provided for a change to the FSAR to remove the description of the Mechanical Environmental Qualification (MEQ) Program for safety-related mechanical and environmental augmented quality equipment. This change affects only the documentation process for MEQ Program equipment located in a harsh environment.

It was concluded from the safety evaluation that the suitability of equipment to design requirements, including environmental conditions, will continue to documented as part of engineering processes. Elimination of the MEQ Program will not alter the equipment, equipment function, plant configuration or maintenance and surveillance requirements.

Existing engineering, procurement, maintenance and surveillance processes remain unchanged and will provide assurance that the equipment continues to meet operability requirements during normal operations and accident conditions.

- SCN 96-017 (SE 96-017)

This SCN provided for a change to the FSAR to reflect the elimination of certain response time testing based on NRC endorsement of BWR Owner's Group Licensing Topical Report NEDO-32291, "System Analysis for Elimination of Selected Response Time Requirements," dated December 28, 1994. In addition, the remaining response time test requirements were relocated from the FSAR to the WNP-2 Licensee Controlled Specifications.

It was concluded from the safety evaluation that none of the proposed changes resulted in a physical change or method of operation for any plant components. Through normal instrument loop calibrations, and other logic and system functional tests required by the Technical Specifications, safety system actuations required by transient and accident analyses remain unchanged.

Each of the selected sensor channels using alternative response time testing were reviewed to ensure that a qualitatively-assessed five second response time was consistent with the necessary actuation times specified within the accident analyses. There were no known failure modes that could be detected by response time testing that could also not be detected by other testing required by the Technical Specifications.

The proposed changes would also provide an improvement to plant safety and operation by reducing the time safety systems are unavailable, reducing the potential for safety system actuations, reducing plant shutdown risk, limiting radiation exposure to plant personnel, and eliminating the diversion of key personnel resources to conduct unnecessary testing.

- SCN 96-021 (SE 96-020)

This SCN provided for a change to the FSAR to reflect the use of Option B of 10CFR50, Appendix J, as the basis for the containment leakage testing program. The NRC has approved Option B which allows for implementation of a performance-based containment leakage rate testing program. This FSAR change implements a program which will assure leakage requirements are within current commitments and design basis assumptions.

It was concluded from the safety evaluation that this change would not impact normal plant operation, means of accident mitigation, or physically alter the design of the plant. The SCN was written to assure compliance with 10CFR50, Appendix J, as described in the SER by using the guidelines established in Regulatory Guide 1.163.

The change does not impact the configuration of any structure, system or component or its ability to meet the designed safety function. Each valve and penetration will continue to meet its designed function to isolate and maintain leakage within the specified limits.

- SCN 96-022 (SE 96-026)

This SCN provided for clarification of the as-built configuration of the purge exhaust portion of the control room remote air intake system. The original design comprised two purge exhaust systems consisting of two electro-hydraulically-operated (EHO) isolation valves in series. A subsequent modification changed the design such that, in each purge system, one valve is equipped with an electro-hydraulic operator and is interlocked with its associated remote air intake valve. The other purge valve is maintained open.

It was concluded from the safety evaluation that this activity does not affect the function of these valves. This SCN was written to correct the discrepancy between the FSAR and actual system configuration. The two disabled valves are maintained open by means of an EHO spring or by a split collar and cap screws.

The position of the valves is indicated in the control room and they do not interact or affect any other systems or components. The function of these valves is to remain open during all normal modes of operation and also during accident and post-accident conditions.

- SCN 96-023 (SE 96-043)

This SCN provided for a change to the FSAR to reflect the removal of reference to Reactor Building Electronics Room Air Conditioner RRA-AC-16.

It was concluded from the safety evaluation that removal, or abandonment in place, of the non-safety related (Quality Class II) air conditioning unit would not affect any safety-related process equipment or systems. Removal or abandonment would also not have an impact on any accident analysis.

The air conditioning unit is not used for any safety function. Deactivation of this unit will lessen the heat load internal to the Reactor Building. Deactivation will also reduce electrical loads on the bus from which the unit is powered.

- SCN 96-025 (SE 96-025)

This SCN provided for a change to the FSAR to reflect the deletion of reference to the Offgas Charcoal Vault Refrigeration and HVAC System.

It was concluded from the safety evaluation that accident scenarios do not credit the Offgas Charcoal Vault Refrigeration and HVAC System with any accident mitigation responsibilities or functions. Furthermore, gross failure of the refrigeration system has no safety implications since the charcoal adsorbers can be isolated and the plant can be safely shutdown without the refrigeration system operating.

The system has no interactions with any equipment or systems that act as safeguards or barriers to prevent or mitigate the consequences of an accident. The system was installed to allow for a sightly longer time for select radionuclides in the offgas stream by cooling the charcoal beds.

The Technical Specification limiting dose rate of 332 millicuries/second after 30 minutes decay will be met, with significant margin, without the vault refrigeration system in operation.

- SCN 96-026 (SE 96-047)

This SCN provided for editorial changes to the FSAR and to reflect existing Plant Service Water (TSW) configuration.

It was concluded from the safety evaluation that the changes do not in any way affect the safe operation of the facility. In addition, the intent of the FSAR and basic operation of the TSW System are unaffected by these changes. The TSW System is not required to perform any safety function.

Implementation of this SCN does not result in any physical changes to the plant or any change to design temperatures and pressures used in evaluations of TSW System piping. The changes consist of clarification and correction of current information pertaining to the TSW System description.

- SCN 96-027 (SE 96-019)

This SCN provided for a change to the Fire Protection Program and Emergency Plan to reflect revision of the training requirements and relocation of requirements pertaining to fire brigade training program.

It was concluded from the SCN that the changes to the fire brigade training program would not lower the ability of safety systems to perform during accident conditions or increase safety system challenges without compensating effects. The changes do not reduce the level of overall training on the mitigation of radiological releases.

The proposed changes do not modify plant equipment and would not result in any new failure modes than those previously evaluated. The changes to fire brigade training have no impact on the probability of occurrence of fires, or other design basis events, and continue to ensure that any fires are extinguished in a safe manner.

- **SCN 96-028** (SE 96-057)

  This SCN provided for a change to the FSAR to reflect deletion of reference to the backup diesel drives for the non-safety related motor-driven air compressors in the Emergency Diesel Generator Starting Air Systems (Divisions 1 and 2).

  It was concluded from the safety evaluation that sparing the backup starting air compressor diesel drives would not affect air receiver pressure or the number of starts or starting capability of the emergency diesel generators. The safety-related portions of the starting air system are isolated from the non-safety related portions of the system by means of check valves.

  The assumed malfunction evaluated previously is the failure of an emergency diesel generator to start and supply power to the critical electrical buses. Therefore, the consequences of a malfunction of any equipment in the diesel starting air system remains unchanged by sparing the backup starting air compressor diesel drives.

- **SCN 96-029** (SE 96-042)

  This SCN provided for revision of the Offsite Dose Calculation Manual (ODCM) description of the Reactor Building Effluent Monitoring System. The description of the main plant vent intermediate and high range detectors was deleted.

  It was concluded from the safety evaluation that detectors PRM-RE-1B and PRM-RE-1C are important to safety Regulatory Guide 1.97 monitors. However, the description of the detectors was not required to be included in the ODCM.

  The elimination of these monitors from the ODCM does not alter any design requirements or methods of operation for the components.

- **SCN 96-033** (SE 96-049)

  This SCN provided for a change to the FSAR to remove unnecessary detailed information pertaining to theory of operation, provide additional clarification where necessary, and to reflect testing results for the low, intermediate and high range detectors in the Reactor Building Noble Gas Effluent Monitoring System.

  It was concluded from the safety evaluation that the radiation monitor function is unaffected by this change. The monitors continue to provide sufficient sensitivity to meet accident monitoring requirements for all calculated maximum accident releases and allow for accurate offsite dose predictions and associated operator decisions.

The tested span for the low range detector is within the range requirements necessary for ODCM monitoring. No physical plant equipment changes were made as a result of this FSAR revision. In addition, this change has no impact on any safety-related or important to safety component function or method of operation.

- **SCN 96-038** (SE 96-065)

This SCN provided for a change to the FSAR to reflect deletion of a reference to Standby Liquid Control (SLC) System minimum area temperature because the SLC tank has safety-related heaters which keep the boron solution above saturation temperature. Furthermore, the piping from the SLC storage tank to the pump suction valves is heat traced and insulated.

It was concluded from the safety evaluation that the sodium pentaborate will be maintained above the saturation temperature. The SLC System is a backup reactivity control system to the Control Rod Drive System and this change does not involve any physical changes to the plant. The heaters are capable of maintaining the boron solution within design temperature ratings to ensure complete solubility of the solution.

- **SCN 96-042** (SE 96-052)

This SCN provided for a change to the Fire Protection Program to reflect the establishment of a new fire barrier operability category, "Operable but Nonconforming," which would result in shiftly (once per 12 hours) fire tours. In addition, this SCN allows for changes to the FSAR for not requiring fire tours in the Main Control Room. The bases for fire-rated assemblies was also clarified.

It was concluded from the safety evaluation that the changes would not lower the ability of safety systems to perform during accident conditions or increase safety system challenges without compensating effects. The changes do not reduce the level of overall training on the mitigation of radiological releases.

The proposed changes would not result in any new failure modes than previously evaluated. The changes have no impact on the probability of occurrence of fires, or other deign basis events, and continue to ensure that any fires are extinguished in a safe manner.

Crediting the Main Control Room Operators to perform the function of a fire tour is acceptable. The control room is continuously occupied and cognizant operators have the capability to detect power generation control complex and control room fires in the incipient stages.

2.7.4 Problem Evaluations

This section contains information pertaining to Problem Evaluation Requests (PERs) and is included pursuant to 10CFR50.59.

- **PER 296-0215** (SE 96-021)

  This PER documented a situation where it was noted that an inflatable seal on Equipment Hatch MT-DOOR-A2 in the Reactor Building railroad crane bay has never been used. This door is part of secondary containment when Reactor Building Railroad Crane Bay Door R-106 is open. The concern was that the inflatable seal may have been required for secondary containment integrity. The disposition of this problem was "permanent accept-as-is."

  It was concluded from the safety evaluation that the seal is not needed on MT-DOOR-A2. The door is a gasketed, airtight and water resistant component capable of withstanding maximum hydrostatic pressures in the event of a pipe rupture. Secondary containment was tested with Door R-106 open and then closed. Leakage rates were approximately the same for both conditions and were within required limits.

  The hatch is interlocked with Door R-106 such that both doors can not be open at the same time to preserve secondary containment integrity. The situation as described on this PER has no impact on the interlock function. It was also concluded that the inflatable seal was not required for flooding considerations.

- **PER 296-0273** (SE 96-036)

  This PER documented a situation where cracks were discovered in some of the plastic lugs molded into the relay bases and relay terminal bases (relay sockets) of certain "plug-in" type relay assemblies manufactured by ASEA. Some spalling was also noted around the mounting screws. The population consisted of seismic Category I relays used in safety-related applications. The disposition of this problem was "permanent accept-as-is."

  It was concluded from the safety evaluation that the relays were demonstrated by test and analysis to be capable of performing their intended safety function with the cracks/damage present. The seismic testing exceeded the required acceleration levels by a significant safety factor. The problems identified would not cause the relays to fail during any plant mode or transient.

Based on an evaluation of relay seismic test report data, it was determined that the fragility levels used in the seismic tests bounded the required accelerations for all the locations where the identified relays are installed. Furthermore, it was concluded that adequate margin exists to conclude that the cracks/damage in the terminal base plastic of the relays was not detrimental to operation during a seismic event.

- **PER 296-0276**

(SE 96-045)

This PER documented a situation where it was observed during inspection of the jet pumps that set screws on Jet Pump No. 18 were not making contact with the inlet mixer to provide for three-point stabilization. A 10-mil gap was observed between the set screws of the restrainer brackets and the inlet mixer. The jet pump design and installation specifications required that there be no gaps at these points. The disposition of this problem was "permanent accept-as-is."

It was concluded from the safety evaluation and a General Electric vibration analysis that gaps of up 12 mils were acceptable. This was based on operating experience of other BWR 5 plants and the determination that, as long as in-vessel inspections at succeeding outages show that the set screw gap is at 12 mils or less, continued operation is acceptable. The function of the jet pumps would not be adversely affected when the gaps are maintained at this limit.

Based on set screw gap vibration analysis and General Electric experience, operation with a gap of 10 mils during one cycle will not affect jet pump structural integrity. Additional operational cycles may also continue with the gaps present as long as the width is less than 12 mils.

(SE 96-046)

This PER also documented a situation where damage was observed on a Jet Pump No. 3 inlet mixer wedge and its associated restrainer bracket wear pad. This problem was discovered during jet pump inspection efforts. The disposition of this problem was "interim accept-as-is."

It was concluded from the safety evaluation that the worn inlet wedge and associated wear pad provide a fixed restraining point for the inlet mixer. It was also concluded that the extent of the damage was not sufficient to cause a failure of either the inlet mixer wedge or restrainer bracket wear pad.

Installation of the restrainer bracket wedge has restored the three-point restraint required to stabilize the inlet mixer and prevent further damage to the inlet mixer wedge. Accordingly, it was concluded from evaluation that the current extent of the degradation would allow operation for an additional cycle.

- **PER 296-0278** (SE 96-048)

This PER described a situation where broken and missing anodes on cooling coils were discovered during cleaning of the Standby Service Water (SSW) System, Loop B, room coolers. The anodes were originally supplied to minimize the potential for corrosion, given the water chemistry specified at the time of coil purchase. The disposition of this problem was "re-work."

It was concluded from the safety evaluation that the presence or absence of anodes has no effect on the ability of the SSW System to meet its cooling requirements. The status of the anodes in the SSW cooling coils does not impact the ability of the system and associated components to mitigate the effects of design basis accidents or support safe shutdown of the plant.

Flow to cooling coil tubes continues around the anode, so no significant flow reduction would be expected to occur if the anode was separated from the cap support. Flow through the cooling coil is adequate for any anode status (installed, removed or damaged).

- **PER 296-0360** (SE 95-102-01)

This PER described a situation where a loose jet pump sensing line clamp was noted at location 10C. A 30-mil gap was observed at the top of the clamp and was due to a circumferential weld being at the location of the clamp. The disposition of this problem was "permanent accept-as-is."

It was concluded from the safety evaluation that clamp integrity was proven through the use of crimp and shaker tools. The clamp was confirmed to be tightly installed and no loosening due to vibration would be expected.

Based on evaluation, it was concluded that the as-installed configuration of the clamp would not impact the lost parts analysis. It was also determined that there would be no impact on core re-flood analysis capability or emergency core cooling system performance.

- **PER 296-0436** (SE 96-051)

This PER described a situation where direct bridging circuit separation discrepancies were identified involving annunciator system field common circuits for cables. The cables bridged directly between Division 1 and Division 2 raceways. The disposition of this problem was "interim accept-as-is." The WNP-2 Electrical Separation Criteria Document (DRD 201) was revised to provide clarification for allowing low energy direct circuit bridges.

It was concluded from the safety evaluation that allowing non-class 1E, low energy (instrument and control) cables to bridge directly between redundant raceways would be acceptable when certain acceptable conditions are met.

These changes maintain WNP-2 commitments to single failure criteria required by 10CFR50, Criterion 17, for safety-related electric power systems and will not increase the probability of occurrence or consequences of any previously evaluated transient. These commitments to electrical separation criteria ensure that, during accident conditions in conjunction with a postulated localized fire, redundant safety functions cannot be affected.

- **PER 296-0438** (SE 96-58)

This PER described a situation where it was noted that it took approximately 15 seconds to withdraw Control Rod Drive (CRD) System Mechanism CRD-DRVE-0627 from position 00 to 02. Normal drive speed is approximately one notch (six inches) every two seconds, with a full stroke in 48 seconds. The drive had to be withdrawn from position 00 using the continuous withdraw command. Beyond position 02, withdrawal speed was within normal limits. Rod insert motion was not affected and was within normal limits. The dispositions of this problem were "interim accept-as-is," and "permanent rework." These classifications were based on continued use of CRD-DRVE-0627 in the degraded condition pending rebuild during the Spring 1997 Maintenance and Refueling Outage.

It was concluded from the safety evaluation that the problem was most likely caused by reversal or degradation of the drive piston drive-down seals (bridge and radial). A degraded or reversed drive-down seal assembly can, over time, impact normal withdrawal movement of the drive. However, this condition would not affect normal insert motion or the safety-related scram function of the drive.

There is also no postulated accident scenario that would require event mitigation through withdrawal of a control rod.

2.7.5 Plant Tests and Experiments

This section contains information pertaining to tests and experiments not described in the FSAR and is included pursuant to 10CFR50.59.

- **PPM 8.9.2** (SE 96-016)

  This procedure provides instructions for monitoring reactor cavity and spent fuel pool temperatures to verify the adequacy of natural circulation as an alternate flow mechanism while in Operational Mode 5 (Refueling), with the reactor cavity flooded and spent fuel pool gates removed.

  It was concluded from the safety evaluation that installation and removal of temperature monitoring equipment in the reactor pressure vessel and spent fuel pool would not introduce a new mechanism for initiation of any evaluated accidents. In addition, this activity would not interfere with the normal sequence of events for mitigating such accidents.

  The monitoring equipment is non-obtrusive and does not rely on plant support equipment other than electrical power to operate. Should electrical power be lost to the temperature monitoring equipment, power could be restored using other power sources or battery-powered components.

2.7.6 Plant Procedure Changes

The section contains information pertaining to Plant Procedure Manual (PPM) changes and is included pursuant to 10CFR50.59.

- **PPM 1.3.10C** (SE 93-093)

  This is a new procedure which describes the administrative program for the control of transient combustible materials and provides for periodic inspection for the accumulation of combustibles. Existing procedural guidance in this area was moved from PPM 1.3.10 to this new procedure.

  It was concluded from the safety evaluation that, although some aspects of the transient combustible program are less restrictive than in previous procedural revisions, the controls are still adequate to ensure that combustibles are limited to the extent practical.

  These changes are not in conflict with previous commitments and do not represent an appreciable reduction in the margin of fire safety. The changes implement a thorough and effective administrative program for control of transient combustibles, while eliminating a few aspects which were deemed to be an overly conservative burden and marginal to safety.

- **PPM 2.2.1A** (SE 96-069)

  This is a new procedure for controlling Reactor Recirculation (RRC) System flow from the local control and diagnostic panel using the adjustable speed drive channels. This procedure was developed to allow for local control in the event of the loss of control of recirculation pump speed in the Main Control Room.

  It was concluded from the safety evaluation that this temporary arrangement does not increase the probability of a recirculation pump transient because the trip function of the Adjustable Speed Drive System has not changed. The recirculation flow control failure is caused by the failure of the master controller. This change is limited to setting the speed limiter to a lower value. The consequences of a flow controller failure would be reduced because the high speed limiter will be set lower that the design value.

  This change also has no impact on recirculation runback due to the loss of a reactor feedwater pump as long as reactor power is maintained less than the equivalent power that could be established on the 108 percent rod line (<65 percent core thermal power). This change does not impose any action or modification that is beyond the designed capability of the Adjustable Speed Drive System or associated components such as the RRC pumps.

- **PPM 2.8.1A** (SE 96-023)

This is a new procedure which provides instructions for Operations personnel to support an outage of the Control Air System (CAS) and Service Air (SA) System for maintenance and inspection purposes. This procedure can be implemented in Operational Mode 4 (Cold Shutdown), Mode 5 (Refueling) or Mode * (Refueling).

It was concluded from the safety evaluation that, during these operational modes, loss of CAS and SA would not result in the loss of any safety-related or important to safety functions. The CAS and SA systems are not safety-related and provide no safety functions. Loss of air to important to safety equipment results in the equipment reverting to the safe condition (position).

It has been determined from failure analyses that no transients beyond those previously evaluated can occur from the loss of either (or both) system. In addition, it was concluded that complete loss of the systems would not affect the results of previously evaluated transients.

- **PPM 2.8.5** (SE 96-018)

This procedure provides for operation of the Fuel Pool Cooling (FPC) and Cleanup System during all operational modes. The procedure was revised to include a section on operation of the Residual Heat Removal (RHR) System in the FPC assist mode when RHR, Loop A, is not available for operation. A section was also added on operation of RHR, Loop A, in a mode to assist in maintaining spent fuel pool temperatures if RHR, Loop B, becomes unavailable during or following a full core off-load.

It was concluded from the safety evaluation that fuel handling and rod withdrawal accident analyses were unaffected by the decay heat removal operating mode of the RHR System. Adequate decay heat removal from the reactor pressure vessel and spent fuel pool would be maintained.

Adequate level would be maintained in the spent fuel pool in accordance with system design, and temperatures would be within allowable limits. Both the FPC and RHR systems would continued to be operated within their design limits.

- **PPM 6.5.19** (SE 96-039)

This procedure provides for resolution of jet pump set screw gap and wedge problems. The procedure was revised to allow for installation of restrainer bracket wedges using a General Electric-approved procedure. The original approach was to install restrainer bracket adjusting screws.

It was concluded from the safety evaluation that installation of the restrainer bracket wedges will replace the function of the restrainer bracket adjusting screws. The wedges will restore the degree of lateral support required for the original design configuration for the jet pump inlet mixer.

The probability of failure of any jet pump component, and subsequent ejections of the jet pump mixer, remains unchanged by installation of the strainer wedges. Because there would be no change in performance of the jet pump design feature, the functional capability of the jet pumps remains unchanged by installation of restrainer bracket wedges.

- **PPM 8.3.339** (SE 96-106)

This procedure provides for testing of the Digital Feedwater Level Control (DFWLC) and Adjustable Speed Drive (ASD) Systems. Included in the procedure was a test to demonstrate the feedwater level control system and recirculation flow run-back feature that would prevent a low vessel water level scram following a single feedwater pump trip during power operation. The safety evaluation was completed to determine the impact of either deferring or not completing this part of the procedure.

It was concluded from the safety evaluation that there were no unacceptable consequences in deferring or not performing this part of the procedure. A comparison of the as-tested ASD run-back rates to previous GE-NE Control System analyses concluded that ASD run-back rates and DFWLC settings are adequate for avoiding a low level scram due to feedwater pump trip during power operation. The DFWLC and ASD recirculation flow control systems are expected to perform, as described in the FSAR, during a feedwater pump trip transient.

Based on system-level test results performed to date, it was determined that an additional integrated test to again confirm system performance or assumptions used in plant transient analyses was unnecessary.

- **PPM 8.3.372** (SE 96-024)

This is a new procedure which provides for motor-operated valve in-situ differential pressure operability testing of High Pressure Core Spray (HPCS) System Injection Valve HPCS-V-4. The procedure was developed to satisfy recommendations contained in the WNP-2 Motor Operated Valve Periodic Verification Plan and NRC Generic Letter 89-13. It may also be used for post-maintenance testing.

The HPCS System provides a mitigating function to maintain reactor inventory or core spray after a loss of coolant accident. The purpose of performing differential pressure testing of HPCS-V-4 is to verify that this valve will properly operate when required to perform its open and close safety functions.

It was concluded from the safety evaluation that the test is conducted during cold shutdown conditions when the HPCS System is not required to be operable and alternate Emergency Core Cooling Systems (ECCS) are available. During the test, reactor grade water from the condensate storage tanks would be injected into the vessel using normal injection alignment when the vessel is vented (head vent valves open or head removed). The automatic interlock for closure of HPCS-V-4 at the Level 8 setpoint (+54.5 inches) would be maintained to prevent potential overpressurization or overfill of the vessel.

It was also concluded that the test conditions established by the procedure would not result in exceeding HPCS System or reactor vessel design or operating limits.

- **PPM 9.3.39** (SE 96-050)

This procedure provides for installation of the cycle-specific input deck and the CREATE base data into the POWERPLEX Core Monitoring Software System (CMSS). The procedure was revised to incorporate the POWERPLEX input database for Cycle 12.

It was concluded from the safety evaluation that the core monitoring system does not interact with plant equipment and will not initiate any previously evaluated accident. The update of the input data base is performed only after completion of detailed engineering calculations.

The POWERPLEX CMSS is used to verify compliance with specific core operating limits. These operating limits are specifically designed to protect against the most limiting accident types. The proposed update will provide the POWERPLEX monitoring system with the ability to use Cycle 12-approved core operating limits for both ABB and Siemens fuel. Therefore, the system will maintain the ability to provide core operating limit evaluations. This activity does not change the method by which the plant is operated in accordance with procedures.

This procedural revision will allow for core monitoring to be performed using appropriate methodology consistent with the Cycle-12 licensing analyses.

- **PPM 10.3.2** (SE 95-043)

This procedure provides instructions for installation and removal of reactor cavity shield plugs and gates. The procedure was revised to allow for removal of the lower set of reactor pressure vessel shield plugs during Operational Mode 3 (Hot Shutdown). In addition, the lift rating of certain slings was increased to 50 tons.

It was concluded from the safety evaluation that the fuel handling accident (dropped fuel bundle into the spent fuel pool) would not be affected by removal of the shield plugs. Removal of the plugs would follow a safe load path which does not include travel over the spent fuel pool. Current procedures allow removal of the top-layer reactor cavity shield plugs while the reactor is critical.

Removing the lower set of reactor cavity shield plugs in Operational Mode 3 would not increase the consequences of a previously evaluated accident. A safety factor of 5:1 would still be maintained by increasing the rating of the lifting slings.

- **PPM 10.24.17** (SE 96-067)

This procedure provides instructions for performing and documenting control rod friction and settle differential pressure testing. The procedure was revised to incorporate enhanced test equipment connection methodology during plant operation. Prior to revision, differential pressure testing equipment was connected to control rod drive insert and withdraw line high point vent valves. This was changed to allow for connection of the testing equipment to manifold test ports on the associated hydraulic control unit.

It was concluded from the safety evaluation that performance of differential pressure testing in accordance with the revised test equipment connection methodology is a controlled evolution specified by the original equipment manufacturer.

This change provides a reliable means to isolate the test equipment from the reactor pressure boundary if necessary. Test connections also restrict the flow path such that any postulated leakage due to breach of the temporary test equipment is bounded by the transient assumed for an instrument line break outside containment.

- **PPM 10.25.155** (SE 96-082)

This procedure provides for monthly inspections of 10CFR50, Appendix R, emergency lighting battery units. The procedure was revised to correct and clarify emergency battery light discharge rates such that the values would be in conformance with the safe shutdown calculation and manufacturer information. In addition, guidance was added pertaining to the posting of approved portable battery-powered lanterns as compensatory measures during discharge testing.

It was concluded from the safety evaluation that there would be no new accident scenarios introduced by these changes. All plant systems and components required to mitigate the consequences of accidents previously evaluated would be unaffected by the changes. Revising the procedure to reflect emergency battery light discharge capacity contained in the shutdown calculation aligns it with plant design. Adding portable lanterns as compensatory measures during discharge testing increases the ability of providing lighting in support of post-fire shutdown operator actions.

The sole purpose of the emergency battery lighting is to provide illumination for personnel exiting the plant during a fire or station blackout.

- **PPM 16.1.1** (SE 96-040)

This procedure provides for channel calibration of Reactor Building Effluent Low Range Radiation Monitor PRM-LCRM-1A. The procedure was revised to relocate the Offsite Dose Calculation Manual (ODCM) high radiation alarm setpoint from the intermediate range detector to the low range detector.

It was concluded from the safety evaluation that the stack monitoring system consists of passive instrumentation and is used to monitor post accident releases from the main plant vent elevated release point. The stack monitoring system does not initiate any accidents.

This change moves the ODCM high radiation alarm function from the intermediate range detector to the more sensitive low range detector, tightens tolerances for loop checks, and allows for adjustment of the normalize potentiometer for testing purposes. The accident monitoring function of the stack monitor would be unaffected by these changes. Failure consequences remain unchanged from the alarm/setpoint relocation. The consequences of a failure of the low range detector would be identical to that of the original intermediate range detector.

2.6.7 Miscellaneous

This section contains information pertaining to other plant activities and is included pursuant to 10CFR50.59.

- **Clearance Order 93-12-0045** (SE 96-059)

  This clearance order allowed for deactivation of Process Sampling System (PSR) Booster Pump PSR-P-26. The pump is used to boost sample flow to Sample Point 26. This temporary change was to remain in place until such time that a permanent resolution to correct sample tube blocking problems can be implemented.

  It was concluded from the safety evaluation that the liquid sampling system has no safety or direct process control functions. Deactivation of the booster pump or isolation of the sample lines would not impact the operation of any equipment important to safety.

  This activity would not impact any system used to ensure the integrity of the reactor coolant pressure boundary, the capability to shutdown the reactor, or the capability to prevent accidents. Chemical analysis of process system liquid will continue through grab sampling at an alternate point (Sample Rack RCC-SR-44).

- **Computer Change Request CCR-TE-95-013** (SE 96-001)

  This change request modified the Siemens Power Corporation POWERPLEX Core Monitoring Software System (CMSS) to allow the MICROBURN code to use the ABB critical power correlation for monitoring ABB fuel. The POWERPLEX CMSS provides core monitoring capabilities by monitoring power distribution.

  It was concluded from the safety evaluation that the POWERPLEX CMSS is not physically connected to any plant safety systems or any other plant systems important to safety, with the exception that power for the computer is from uninteruptible power source IN-1. The POWERPLEX CMSS does not cause any installed plant safety systems to activate. The computer on which the system is installed is Quality Class G hardware and uses a standard Digital VMS operating system.

  There are no environmental or technical qualifications for the computer or its operating system. Since the POWERPLEX CMSS is not physically connected to any plant safety systems, it is not and can not be an initiator for any plant transient or accident. Furthermore, changes to the computer program would not change any operational modes, operating procedures, or method of operating plant equipment.

- **Core Operating Limits Report 96-12:Rev 0** (SE 96-031)

This revision allowed for implementation of the WNP-2, Cycle 12, Core Operating Limits Report (COLR). The proposed activity consisted of operation of the Cycle 12 reload core with core thermal limits which have been developed with NRC-approved methodologies. The thermal limits are specified in the COLR. The Cycle-12 reload core consists of fuel assemblies of the SVEA-96 design, Siemens 9x9-9x design and Siemens 8x8-2 design. The SVEA-96 assemblies are new to WNP-2 with this reload.

It was concluded from the safety evaluation that operation of Cycle 12 within the thermal limits defined in COLR 96-12 does not increase the consequences of the analyzed anticipated operational occurrences or accidents because the mechanical, thermal hydraulic and LOCA design criteria imposed on the fuel to protect it during these events are met.

Analyses of the previously-evaluated accidents and bounding anticipated operational occurrences systematically addressed all fuel characteristics, fuel related equipment malfunctions and operator actions. The depth of these analyses precludes the possibility of an accident which has not been previously evaluated, provided that the linear heat generation rate and other thermal limits as established by the COLR are followed.

- **Fire Protection: Penetration Seals** (SE 96-053)

This activity consisted of the continuous use of fire tours as adequate ongoing compensatory measures for inoperable fire-rated penetration seals.

It was concluded from the safety evaluation that fires are neither initiators nor mitigators of any previously analyzed transients. These tours can decrease the probability of occurrence of plant fires in that they may discover smoldering pre-fire conditions which could be mitigated prior to outbreak.

This activity does not degrade or prevent actions assumed in the accident analysis, adversely affect fission product barriers, alter any assumptions made in evaluating radiological consequences of an accident, or physically modify any plant component or system operation.

- **Fire Protection: Vertical Cable Trays** (SE 96-054)

This activity consisted of the continuous use of fire tours as adequate ongoing compensatory measures for inoperable Thermo-Lag coated vertical cable tray fire breaks.

It was concluded from the safety evaluation that fires are neither initiators nor mitigators of any previously analyzed transients. Fire tours can decrease the probability of occurrence of plant fires in that they may discover smoldering pre-fire conditions which could be mitigated prior to outbreak.

This activity does not degrade or prevent actions assumed in the accident analysis, adversely affect fission product barriers, alter any assumptions made in evaluating radiological consequences of an accident, or physically modify any plant component or system operation.

- **Technical Evaluation Request 96-0009-0** (SE 96-028)

This Technical Evaluation Request provided for the relocation of two Bailey cards and installation of sliding link terminal blocks for Primary Containment Sump Flow Monitoring System and Reactor Water Cleanup System isolation instrumentation (LD-SUM-604 and FDR-SQRT-38).

It was concluded from the safety evaluation that the function of the instruments would not be affected by the change in position or the addition of terminal blocks. There are no transients or accidents that would be affected by this activity. The cards were simply moved to a new location within the same rack and would provide the same functions as the original configuration.

The installation of the terminal blocks with sliding link disconnects allows for system testing without removing any wiring. System interfaces are not changed and design basis requirements for electrical separation, seismic and instrument loop tolerances were maintained.

- **Technical Evaluation Request 96-0127-0** (SE 96-109)

This Technical Evaluation Request provided for the permanent designation of temporary valves and spectacle flanges installed in the water box drain lines to allow for on-line tube plugging in the main condenser.

It was concluded from the safety evaluation that implementation of this change would not adversely impact condensate system piping stress evaluations. The pressure ratings of the current components are within the required tolerances for the system. The condensate system is a Quality Class II, non-safety related system.

The system is not required to perform any safety function such as maintaining the integrity of the reactor coolant pressure boundary or mitigating the consequences of an accident.

- **Technical Evaluation Request 96-0178-0** (SE 96-102)

This Technical Evaluation Request provided for the replacement of several globe and gate pattern valves in the Plant Service Water (TSW) System with ball pattern valves. The valves were replaced to allow for drain water to be routed through the equipment/floor drain system.

It was concluded from the safety evaluation that the replacement ball pattern valves have equal or greater pressure ratings than the existing valves and they weigh less. The TSW is a Quality Class II, non-safety related system. The system is not required to perform any safety function such as maintaining the integrity of the reactor coolant pressure boundary or mitigating the consequences of an accident.

Stress evaluations for small bore piping and valves in these applications would not be adversely impacted by this change.

- **Work Order YS6901** (SE 96-009)

This work order provided for repair of Demineralized Water System Valve DW-V-100/57 and isolation of the component from the system by means of a freeze seal in the horizontal run of piping upstream of the valve.

It was concluded from the safety evaluation that the only credible problem area of concern during isolation of the water supply to DW-V-100/57 would be flooding of the Reactor Building 471' elevation due to failure of the freeze seal. A decrease in reactor coolant inventory would require freeze seal failure, coincident with a Post Accident Sampling System (PASS) sample being drawn from the jet pumps and failure of PASS Check Valve PSR-V-106 in the Demineralized Water System flush line. However, this event has been previously evaluated and documented in the FSAR.

All important to safety equipment affected by a maximum potential flooding event is already identified and included in the plant flooding analysis. Flooding due to failure of the freeze seal would not alter the manner in which equipment is assumed to fail during any flooding event.

- **Work Orders WT5001 and YH1001** (SE 96-037)

These work orders provided for installation of a freeze seal in a Control Rod Drive (CRD) System line upstream of CRD Vent Valve CRD-V-101A to allow for inspection of CRD Insert Isolation Valve CRD-V-101/5027. The proposed activity would be performed with the reactor in cold shutdown, depressurized and all control rods fully inserted.

It was concluded from the safety evaluation that there were no postulated accidents or operational transients associated with a break in the one-inch CRD insert line that could result in an increase in the radiological dose at the site boundary. There were also no postulated accidents or operational transients associated with a break in the line that could result in flooding and impact the functional ability of equipment important to safety.

In was also concluded that the piping was capable of accepting the stresses from the freeze seal. If the pressure integrity of the insert line were to fail due to loss of the freeze seal during the rework or replacement of the valve, no control rod withdrawal would occur.

- **Work Order YT6201** (SE 96-038)

This work order provided for installation of a freeze seal in a Control Rod Drive (CRD) System line upstream of CRD Vent Valve CRD-V-102A to allow for inspection of CRD Withdraw Isolation Valve CRD-V-102/2251. The proposed activity would be performed with the reactor in cold shutdown, depressurized and all control rods fully inserted.

It was concluded from the safety evaluation that there were no postulated accidents or operational transients associated with a break in the one-inch CRD insert line that could result in an increase in the radiological dose at the site boundary. There were also no postulated accidents or operational transients associated with a break in the line that could result in flooding and impact the functional ability of equipment important to safety.

In was also concluded that the piping was capable of accepting the stresses from the freeze seal. If the pressure integrity of the insert line were to fail due to loss of the freeze seal during the rework or replacement of the valve, no control rod withdrawal would occur.

2.8 REPORT OF DIESEL GENERATOR FAILURES

This section contains information pertaining to diesel generator failures, valid and non-valid, and is included pursuant to Technical Specifications 4.8.1.1.3 and 6.9.1.

There was one non-valid failure in 1996. There were no valid load demand failures for the three emergency diesel generators.

The non-valid failure event was documented on Problem Evaluation Request 296-0338 and is described as follows:

- **Identity of Diesel Generator and Date of Failure**

  Division One Emergency Diesel Generator (DG-1): May 8, 1996 (2128 hours)

- **Number Designation of Failure in Last 100 Valid Tests**

  This was the first failure of the last 100 tests. However, this test was determined to be a "non-valid" load demand failure.

- **Cause of Failure**

  During performance of the annual LOOP/LOCA surveillance test for the Division 1 Emergency Diesel Generator, a fuel oil leak developed on the return line to the fuel oil day tank. The leak occurred on a threaded fitting to the flanged connection of the return line.

- **Corrective Measures Taken**

  The unit was shutdown and the fuel line was replaced and leak tested.

  Upon completion of successful repair efforts, the diesel generator was re-started and the 24-hour surveillance test was completed without further incident.

- **Length of Time Diesel Generator Unavailable**

  The Diesel Generator was out of service for approximately one hour. Testing activities resumed at 0507 hours on May 9, 1996.

- **Current Surveillance Interval**

  Thirty one days.

2.9 REGULATORY COMMITMENT CHANGES (NEI PROCESS)

This section contains information pertaining to Regulatory Commitment Changes (RCCs) and is included pursuant to the NEI Guidelines for Management NRC Commitments.

- **RCC-30985-00** (Vital Area Access Controls)

  The original commitment description is, "The Security Supervisor responsible for categorizing materials entering the protected area will be present initially to ensure proper vital area access controls are in position anytime (door) R-106 is opened to allow access." This commitment was made in response to Security Event Report 87-004.

  This commitment was deleted. The basis for deletion is that a current procedure requires that, prior to allowing personnel access into any vital area, access authorization for the area is verified by either a review of an access authorization list or through communications with the Central Alarm Station. This alternate method complies with 10CFR73.55(g)(1) and 10CFR73.55(d)(7)(B).

  There is no change to the protection level for personnel to ensure a system, structure or component is capable of performing its function. Access authorization continues to be verified prior to personnel being allowed entry into a vital area.

- **RCC-106987-00** (Crane Stops)

  The original commitment description is, "Place a physical stop on the monorail supporting MT-HOI-6 to prevent moving RHR - Loop A or Loop B components over the equipment left in service." This commitment was made in response to NUREG-0612 [Letter GO2-83-614, dated July 13, 1983, GC Sorensen (SS) to A Schwencer (NRC), "Response to NUREG-0612 -- Phase II, Control of Heavy Loads; Submittal of" (sic)].

  This commitment was deleted. The basis for deletion is guidance contained in Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," dated June 28, 1995. This Generic Letter provided relief to certain previous requirements pertaining to heavy loads. This commitment was within the scope of those requirements that were allowed to be eliminated.

  Current procedures are in place to meet the requirements of Phase I and ensure the safe operation of this hoist.

- **RCC-107116-00** (Positioning of Cranes and Hoists)

The original commitment description is, "Certain cranes and hoists will be locked out in a safe position and not placed into use until the equipment they service has been declared inoperable per the Plant Technical Specifications." This commitment was made in response to NUREG-0612 [Letter GO2-82-824, dated October 4, 1982, GD Bouchey (SS) to A Schwencer (NRC), "Response to NUREG-0612, Control of Heavy Loads, Revision 1; Submittal of" (sic)].

This commitment was deleted. The basis for deletion is guidance contained in Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," dated June 28, 1995. This Generic Letter provided relief to certain previous requirements pertaining to heavy loads. This commitment was within the scope of those requirements that were allowed to be eliminated.

Current procedures are in place to meet the requirements of Phase I and ensure the safe operation of cranes and hoists.

- **RCC-107245-00** (Crane Operator Training)

The original commitment description is, "There are no exceptions taken to ANSI B30.2-1976 with respect to operator training, qualification and conduct. Per plant procedures, this type of training is documented and records of such training are maintained 'current' by the training staff. Recertification is required every three (3) years unless the operator has not operated the crane during the year and in that case the operator must be recertified before operating the unit." This commitment was made in response to NUREG-0612 [Letter GO2-82-824, dated October 4, 1982, GD Bouchey (SS) to A Schwencer (NRC), "Response to NUREG-0612, Control of Heavy Loads, Revision 1; Submittal of" (sic)].

This commitment was revised to, "There are no exceptions taken to ANSI B30.2-1976 with respect to operator training, qualification and conduct. Per plant procedures, this type of training is documented and records of such training are maintained." The basis for revision is that "recertification every three years unless the operator has not operated the crane during the year and in that case the operator must be recertified before operating the unit" is not required by ANSI B30.2-1976. In addition, documentation is retained in a "current" status and is available for review at the Supply System. This approach to document maintenance is consistent with previous guidance issued in this area.

Recertifications are currently evaluated on a case-by-case basis and could depend on several factors such as job complexity or past performance of the crane operator.

- **RCC-131354-00** (Scram Time Testing)

The original commitment description is, "Perform scram time testing once per 60 days on a reference sample of rods having Viton SSPVs consisting of 5% but not less than 5 rods." This commitment was made in response to BWROG SSPV testing recommendations (Letter GO2-96-078, dated April 5, 1996, JV Parrish (SS) to AC Thadani (NRC), "CRD SSPVs with Viton Internals").

This commitment was revised to, "Perform scram time testing once per 60 days, plus or minus 25%, on a reference sample of rods having Viton SSPVs consisting of 5% but not less than 5 rods." The basis for revision is that the observed rate of change in the performance characteristic of the Viton SSPVs is slow enough to allow adequate response time to an undesirable degraded condition with a 25 percent increase in the testing interval. Furthermore, the 25 percent tolerance is consistent with Technical Specification 4.0.2.

A tolerance was not specified in the original commitment and this revision simply clarifies the intended testing interval tolerance.

- **RCC-135865-00** (Procedure Review Committee)

The original commitment description is, "A POC procedure review committee has been established as part of the procedure change management process to perform additional procedure review prior to general POC member review." This commitment was made in response to NRC Inspection Report 93-45 (Letter GO2-94-026, dated January 28, 1994, JV Parrish (SS) to NRC, "NRC Inspection Report 93-45 Response to Notice of Violation").

This commitment was deleted. The basis for deletion is that, as part of a procedure upgrade project, the Supply System transitioned from having department procedure coordinators and procedure reviews performed by selected reviewers, to procedure sponsors and qualified procedure reviewers. Procedure sponsors are identified as the procedure owner (i.e., most knowledgeable of the procedure). Qualified procedure reviewers are trained and qualified to perform procedure reviews. Procedure review expectations have also been clearly defined.

Monitoring of procedure sponsor and qualified procedure reviewer performance for procedure revisions and reviews showed improvement to indicate that procedure review committee reviews were no longer necessary. This monitoring was performed by the procedure review subcommittee.