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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

P.O. Box 968 • Richland, Washington 99352-0968

February 26, 1998
GO2-98-038

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 1997

The annual operating report for calendar year 1997 is attached. If you have any questions or desire additional information pertaining to this report, please contact either me or PJ Inserra at (509) 377-4147.

Respectfully,

Mathew Otto Hammond for

RL Webring
Vice President, Operations Support/PIO
Mail Drop PE08

JDA/lw

Attachment

cc: EW Merschoff, NRC - Region IV
KE Perkins, Jr. NRC - Region IV, WCFO
C Poslusny, Jr. NRC - NRR
REIRS Project Manager, NRC - NRR
NRC Resident Inspector - 927N
DL Williams - BPA - 399
PD Robinson - Winston & Strawn

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WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1997

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

9803060176

WNP-2 1997
Annual Operating Report

Table of Contents

1.0 Introduction

The 1997 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.

Operational Summary

On March 27, 1997 WNP-2 had operated a record 270 days when the plant was scheduled to be taken off-line the same day at the request of the Bonneville Power Administration, customer for WNP-2 electricity, due to an abundance of relatively inexpensive power from the Federal Columbia River Power System.

As part of the scheduled shutdown sequence, post-modification testing of the digital feedwater level control and reactor recirculation adjustable speed drive systems was being conducted to demonstrate that the level control system and recirculation flow runback feature would prevent a reactor scram following the trip of a single feedwater pump during power operation.

After the trip of a feedwater pump, a reactor recirculation pump speed runback to 27 Hz was observed as expected. Following this runback, a second unexpected runback to 15 Hz occurred. The second runback to 15 Hz was caused by a reactor recirculation pump differential cavitation interlock condition existing for greater than 15 seconds, and appeared to have placed the reactor into Region A of the power-to-flow map. (It was later determined by follow-up engineering analyses that Region A had not been entered.)

Accordingly, operators responded conservatively by immediately scramming the reactor and maneuvering the plant to a safe shutdown condition in accordance with procedures. The plant was maintained in a shutdown condition until the scheduled April 1997 start date for the annual maintenance and refueling outage. The 270 day run surpassed the plant's previous continuous record for operation of 257 days, which was set in 1994.

On April 18, 1997 the plant officially entered the 1997 Maintenance and Refueling Outage (R-12) as scheduled. The plant ended the annual outage on June 16, 1997 and the Bonneville Power Administration requested that WNP-2 remain in a shutdown condition due to an abundance of water from the region's hydroelectric projects. The reactor reached criticality on July 4, 1997 and began power ascension. On July 10, 1997 the plant returned to power operation and remained on-line for the remainder of the year.

Refueling Outage Summary

The twelfth refueling outage was successfully completed during 1997. Significant planned and emergent activities included:

- Replacement of 112 fuel assemblies.
- Removal of Asea Brown-Boveri (ABB) debris filters on all ABB fuel assemblies. (WNP-2 was the only plant plant where this evolution had been performed.)
- Refurbishment of nine main steam relief valves.
- Replacement of the rotor on one of the two reactor feedwater pumps.
- Inspection of the low pressure turbine.
- Replacement of 18 control rod drive mechanisms.
- Inspection of reactor recirculation jet pump riser welds.
- Cleaning and inspection of the suppression pool.
- Review of the reactor recirculation differential cavitation interlock setpoint and methodology to determine if any runback value or cavitation logic changes were necessary. As a result of this review, changes were made to the differential temperature cavitation interlock setpoint and differential time delay to runback.

The total outage dose was 195 person-rem, 14 person-rem under the stretch goal and 65 person-rem under the original goal. This was the first outage where exposure was under 200 person-rem.

Other Highlights

To demonstrate that the feedwater level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation, a test was successfully completed on July 23, 1997 when the plant returned to stable, full power operation following the R-12 Maintenance and Refueling Outage. The systems responded as designed.

During 1997, a new power generation record was set for the month of August. The gross generation for the month was 846,420 megawatt-hours and net generation equaled 813,616 megawatt-hours. The previous August record of 823,450 megawatt-hours gross and 791,331 megawatt-hours net occurred in 1994.

The Bonneville Power Administration, on several occasions throughout the year, also requested that WNP-2 reduce power levels so that the federal power marketing agency could maximize generating capability from the region's hydroelectric projects.

2.0 Reports

The reports in this section are provided pursuant to: 1) the requirements of Technical Specification 5.6.1, "Occupational Radiation Exposure Report," 2) the requirements of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," 3) the requirements of 10 CFR 50.59, "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 Specification 5.6.1 requires that the following report be submitted in accordance with 10 CFR 50.4 by April 30 of each year:

> A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was performed, receiving an annual deep dose equivalent of greater than 100 mrems and the associated collective deep dose equivalent (reported in man-rem) according to work and job functions [e.g., reactor operations surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling]. This tabulation supplements the requirements of 10 CFR 20.2206. The dose assessments to various duty functions may be estimated based on electronic or pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totaling less than 20 percent of the individual total dose need not be accounted for. In aggregate, at least 80 percent of the whole body dose received from external sources should be assigned to specific major work functions.

Regulation 10 CFR 50.46 requires that, for each (non-significant) change to or error discovered in an acceptable Emergency Core Cooling System (ECCS) cooling performance evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in 10 CFR 50.4.

Regulation 10 CFR 50.59 requires that licensees submit, as specified in 10 CFR 50.4, 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 the NEI decision criteria, the guidance specifies periodic staff notification, either annually or along with the FSAR updates as required by 10 CFR 50.71(e).

2.1 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 1997

- At the beginning of the month, the plant was at 100 percent full power operation. On January 4, 1997 a phase locked loop load fault occurred on a reactor recirculation system adjustable speed drive channel. The fault caused the drive to trip and initiated a speed reduction to 15 Hz for reactor recirculation system pump RRC-P-1A. Reactor power was reduced to 89 percent. Following extensive troubleshooting efforts, the reason for the load fault was indeterminate.[1]

- On January 7, 1997 reactor power was lowered to 54 percent to install a special file in the adjustable speed drive computer to monitor the phase locked loop load fault trip circuit and to conduct additional troubleshooting activities. On January 8, 1997 troubleshooting efforts were completed and the adjustable speed drive was successfully restarted. Power was then increased to 75 percent for turbine valve testing. Following the testing, the plant returned to 100 percent power.

- On January 8, 1997 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 70 percent. Following economic dispatch, the plant returned to full power operation on January 9, 1997.

- On January 9, 1997 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 70 percent. Following economic dispatch, the plant returned to full power operation on January 10, 1997.

- On January 10, 1997 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 70 percent. Following economic dispatch, the plant returned to full power operation on January 13, 1997.

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1. Extensive troubleshooting, failure analyses and reviews of the results by both Supply System and General Electric personnel determined several months later that the reason for the fault was due to a defect on the NFSC logic card. Variance in card capacitor values had introduced an abnormal signal response from the NFSC card. Accordingly, logic cards were replaced.

- On January 15, 1997 plant operators were responding to average power range monitor upscale alarms, followed by a reactor recirculation adjustable speed drive trouble alarm, and noticed that reactor power was fluctuating up to 20 percent. The operators quickly identified the cause of the power anomaly as oscillating reactor recirculation system pump RRC-P-1A flow. Pump flow was varying between 50,000 gallons per minute and 20,000 gallons per minute. Pump RRC-P-1A was subsequently tripped and operators transitioned the plant for single loop operation. The total elapsed time from initiation of the event to tripping of RRC-P-1A was approximately 50 seconds.

  The reason for the oscillating reactor recirculation system flow was due to a malfunction in the reactor recirculation adjustable speed drive system. The reason for the malfunction was indeterminate. Detailed troubleshooting efforts revealed no known mechanical or electrical failures. It was concluded that the only credible cause for the control system behavior was electronic noise introduced into the adjustable speed drive channel control system.

  Instrumentation was installed on the reactor recirculation system adjustable speed drives in an attempt to trace the cause of the noise. The plant remained in single loop operation for the remainder of the month.

- On January 30, 1997 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time power was further reduced to about 55 percent. The plant remained at this power level for the remainder of the month.

February 1997

- At the beginning of the month, the plant remained in single loop operation. Power level was limited to 55 percent for load following at the request of the Bonneville Power Administration. On February 4, 1997 power was increased to the single loop maximum of 67 percent.

- On February 7, 1997, following extensive troubleshooting efforts and the replacement of logic cards on the adjustable speed drive controls, power was reduced to approximately 30 percent in preparation to restart pump RRC-P-1A.

- On February 9, 1997 pump RRC-P-1A was restarted and plant operators commenced raising power. Adjustable speed drive channel 1A1 tripped and, following discussions with Engineering, operators resumed power ascension with RRC-P-1A running at reduced frequency on a single channel. Troubleshooting efforts continued on the adjustable speed drive channel.

- On February 10, 1997 plant operators stopped power ascension at 90 percent and began a downpower to 70 percent for load following. at the request of the Bonneville Power Administration.

- On February 11, 1997 power level had reached 94 percent during ascension following economic dispatch when pump RRC-P-1A tripped off-line. Plant operators transitioned the plant for single loop operation. Power level was reduced to 65 percent.

- Pump RRC-P-1A tripped during adjustable speed drive channel 1A1 troubleshooting efforts. During the effort, an electrical arc was drawn due to the incorrect connection of an oscilloscope. This resulted in the trip of the channel. The plant remained at 65 percent power until February 15, 1997.

- On February 15, 1997, following extensive troubleshooting efforts, power was reduced to approximately 30 percent in preparation to restart pump RRC-P-1A. The reactor recirculation pump was successfully re-started and the plant returned to two loop operation.

- On February 16, 1997 the plant reached 70 percent power and was held at that level due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On February 17, 1997 the plant began a pattern of 90 percent power during the day and reduced power at night (approximately 55 percent) due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On February 20, 1997 the plant began a pattern of 70 percent power during the day and reduced power at night (approximately 55 percent) due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On February 23, 1997 operators commenced a power reduction to 25 percent in preparation to restart reactor recirculation adjustable speed drive channel 1A1. On February 24, 1997 efforts were undertaken to remove, test and replace several circuit boards in the 1A1 channel. An additional monitor was also installed. Following these efforts, the adjustable speed drive channel was successfully re-started. Plant operators maneuvered the plant to approximately 90 percent power. All adjustable speed drive channels were in operation.

- On February 25, 1997 the plant began a pattern of 100 percent power during the day and reduced power at night (approximately 90 percent) due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On February 28, 1997 a downpower from 100 percent was commenced due to economic dispatch (load following) at the request of the Bonneville Power Administration. The plant ended the month at approximately 75 percent power.

March 1997

- 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 reduced from approximately 75 percent to about 55 percent.

- From March 3, 1997 to March 6, 1997 the plant followed a pattern of full power during the day and reduced power at night (approximately 70 percent) due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On March 6, 1997 the plant returned to 100 percent full power operation and operated at or near that level until March 13, 1997.

- From March 13, 1997 to March 20, 1997 the plant followed a pattern of full power during the day and reduced power at night (approximately 70 percent) due to economic dispatch (load following) at the request of the Bonneville Power Administration.

- On March 20, 1997 with the plant operating at 100 percent power, the NRC Staff notified the Supply System that WNP-2 Technical Specification response time testing surveillance requirements were not being met for certain instruments in the reactor protection, primary containment and emergency core cooling systems. Accordingly, the affected equipment was declared inoperable and a plant shutdown was initiated as required by the Technical Specifications (Reference LER 97-003-00).

  On the same day, plant power had been reduced to 40 percent when the Supply System received enforcement discretion for an exemption from the Technical Specification requirements pertaining to response time testing. Power was then increased to 55 percent and held at that level for economic dispatch (load following) at the request of the Bonneville Power Administration. On March 25, 1997 reactor power was increased to 70 percent.

- On March 27, 1997 WNP-2 had operated a record 270 days when the plant was scheduled to be taken off-line the same day at the request of the Bonneville Power Administration due to an abundance of relatively inexpensive power from the Federal Columbia River Power System.

  As part of the scheduled shutdown sequence, post-modification testing of the digital feedwater level control and reactor recirculation adjustable speed drive systems was being conducted to demonstrate that the level control system and recirculation flow runback feature would prevent a reactor scram following the trip of a single feedwater pump during power operation. In preparation for the test, reactor power was increased to 96 percent.

After the trip of a feedwater pump, a reactor recirculation pump speed runback to 27 Hz was observed as expected. Following this runback, a second unexpected runback to 15 Hz occurred. The second runback to 15 Hz was caused by a reactor recirculation pump differential cavitation interlock condition existing for greater than 15 seconds, and appeared to have placed the reactor into Region A of the power-to-flow map. (It was later determined by follow-up engineering analyses that Region A had not been entered.) Accordingly, operators responded conservatively by immediately scramming the reactor and maneuvering the plant to a safe shutdown condition in accordance with procedures. At the time of the manual scram, plant power level was at 46 percent (Reference LERs 97-004-00 and 97-004-01).

- The 270 day run surpassed the plant's previous continuous record for operation of 257 days, which was set in 1994.

- The plant was maintained in a shutdown condition until the scheduled April 1997 start date for the annual maintenance and refueling outage.

April 1997

- At the beginning of the month the plant continued to be in a reserve shutdown condition. On April 18, 1997 the plant officially entered the annual maintenance and refueling outage.

May 1997

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

June 1997

- The plant officially ended the annual maintenance and refueling outage on June 16, 1997.

- The Bonneville Power Administration requested that WNP-2 remain in a reserve shutdown condition for the remainder of the month due to an abundance of water from the region's hydroelectric projects.

July 1997

- The plant remained in a reserve shutdown condition until July 10, 1997. Following startup and power ascension testing, the plant returned to 100 percent full power operation and remained near or at that level until July 23, 1997.

- To demonstrate that the feedwater level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation, a test was successfully completed on July 23, 1997 when the plant returned to stable, full power operation following the R-12 Maintenance and Refueling Outage. The systems responded as designed. During this evolution, power level was reduced to 58 percent.

- On about July 24, 1997 the plant returned to 100 percent full power operation and remained at or near that level for the rest of the month.

August 1997

- The plant entered the month at 100 percent power and operated at or near that level until August 8, 1997. From August 8 to August 11, 1997 the plant was on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time, power level was approximately 90 percent.

- On August 12, 1997 the plant resumed 100 percent power operation. Following return to full power operation, a reactor recirculation system adjustable speed drive channel tripped during a glycol filter change effort. As a result, power was reduced to 67 percent to restore the channel to service. Following successful restoration efforts, power level was raised to 100 percent and remained at that level until near the end of the month.

- On August 29, 1997 the plant was placed on economic dispatch (load following) at the request of the Bonneville Power Administration. The plant remained at 80 percent power for the remainder of the month.

September 1997

- The plant entered the month at about 80 percent power for economic dispatch (load following) at the request of the Bonneville Power Administration. During the morning of September 1, 1997 power level was raised to 100 percent. The plant remained at or near 100 percent until September 19, 1997.

- From September 19, 1997 to September 22, 1997 the plant was on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time, power generation was approximately 80 percent.

- On September 23, 1997 the plant resumed 100 percent power operation until September 26, 1997. From September 26, 1997 to September 29, 1997 the plant was on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time, power generation was approximately 80 percent.

- On September 30, 1997 the plant returned to 100 percent full power operation.

October 1997

- From October 1, 1997 through October 10, 1997 the plant was on economic dispatch (load following) at the request of the Bonneville Power Administration. During this time, power generation was approximately 80 percent.

- On October 11, 1997 power was reduced to 45 percent for a steam leak repair and reactor feedwater pump troubleshooting. Following steam leak repair efforts, power was increased to 65 percent. The plant was held at 65 percent power for economic dispatch (load following) at the request of the Bonneville Power Administration until October 15, 1997.

- From October 16, 1997 to October 19, 1997 with the plant at 65 percent power, work resumed on the reactor feedwater pump drive turbine control system. One feedwater pump at a time was removed from service.

- Following successful repair efforts on October 19, 1997 power level was raised to 90 percent to perform gain adjustments and testing of the reactor feedwater pump drive turbines. On October 20, 1997 power was raised to 100 percent.

- On October 25, 1997 power was reduced to 79 percent for condensate filter demineralizer maintenance. Following maintenance, power was raised to 100 percent and the plant operated at that level for the remainder of the month.

November 1997

- The plant entered the month at 100 percent power. 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 1997

- The plant entered the month at 100 percent power. 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.

2.2 Significant Maintenance Performed on Safety-Related Equipment

This section contains brief descriptions of major, safety-related maintenance performed during outages or power reductions and is included pursuant to Regulatory Guide 1.16, Section C.1.b(2)(e).

- **Containment System**

  The drywell was inspected. A walkdown of 100 percent of the insulation was conducted, insulation repairs and replacements were made, and coating was applied to piping.

  The wetwell was inspected, cleaned and vacuumed. Design inspections were also performed on all of the emergency core cooling system suction strainers to evaluate underwater installation of the new strainers, which is planned for the Spring 1998 maintenance and refueling outage.

  These efforts, in part, were conducted in support of addressing issues pertaining to NRC Bulletin 95-02, "Unexpected Clogging of Residual Heat Removal (RHR) Pump Strainer while Operating in Suppression Pool Mode," and NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors."

- **Control Rod Drive System**

  Two accumulator tanks were replaced as part of an ongoing program to address problems with deterioration of the chrome surface within carbon steel accumulators. The new accumulators are stainless steel.

  Eighteen control rod drive mechanisms were replaced as part of an ongoing corrective maintenance program.

  The scram solenoid pilot valves were replaced on 77 hydraulic control units to complete the program to address the relatively short lifespan of 1994-vintage buna-n diaphragms. Valves containing buna-n diaphragms were replaced with new valves containing viton diaphragms. This is expected to increase the lifespan of the valves from two-to-three years to 10-to-15 years.

- **Electrical Power System**

  Six Division 2 emergency diesel generator starting air pressure switches were replaced with low deadband components. The existing pressure switches had a large deadband, which impacted the ability of the components in meeting setpoint calculation tolerances.

This change will allow for the switches to be reset within calculated permissible ranges. The change will also help ensure that air receiver pressure remains above the minimum required to provide five starts of the emergency diesel generator.

- Main Steam System

Nine safety relief valves were replaced as part of an ongoing corrective maintenance program to minimize leakage into the wetwell. Leakage into the wetwell presents unnecessary operational challenges such as water level and air space temperature increases.

Drain valves were installed on all eight main steam isolation valve accumulators. This will allow for venting of the pneumatic supplies to the valves for leak testing and provide for periodic blowdown of the accumulators to remove potential debris or moisture which may have migrated to the accumulators.

Main steam system valve MS-V-5 and the associated motor operator were removed from the reactor head vent system that is connected to Loop A of the main steam system in the drywell. There was a history of leakage through the valve packing and uncontrolled leakage through the packing had the potential to result in unacceptable drywell unidentified leakage. Previous efforts to correct the leakage were unsuccessful and the 250-lb valve was replaced with two-inch, Schedule 160, carbon steel piping.

- Neutron Monitoring System

The four source range monitoring and the eight intermediate range monitoring system drives and motor modules were inspected, cleaned and lubricated as part of an ongoing corrective maintenance program.

- Reactor Core Isolation Cooling System

The sealing mechanism for reactor core isolation cooling system flex wedge valve RCIC-V-13 was modified to provide bonnet venting to the high pressure piping. The concern was of rapid reactor depressurization prior to the open signal for reactor core isolation cooling injection. The sealing mechanism was changed such that it seals on only one of the two wedge seats through the use of a bypass port.

This action was a commitment made in response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves."

- **Reactor Recirculation System**

Restrainer wedges were installed on jet pumps MS-JP-14 and MS-JP-18. The restrainer wedges will reestablish the three-point contact between the inlet mixer and the restrainer bracket for each of these jet pumps to resolve an issue of gaps discovered between the set screws of the restrainer brackets and the inlet mixers for the pumps. This change provides the lateral restraint function originally provided by the jet pump inlet mixer setscrews.

The inlet mixer assemblies on two other jet pumps were detensioned, re-seated and re-tensioned, which corrected the set screw gap problem.

- **Residual Heat Removal System**

The sealing mechanism for residual heat removal system flex wedge valve RHR-V-42B was modified to provide bonnet venting to the high pressure piping. The concern was of rapid reactor depressurization prior to the open signal for low pressure coolant injection. The sealing mechanism was changed such that it seals on only one of the two wedge seats through the use of a bypass port.

This action was a commitment made in response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves."

- **Standby Service Water System**

Drain ports were installed on the spray pond return lines for both loops to provide for freeze protection.

Orifices were installed on the spray bypass line to minimize cavitation in the standby service water return lines.

2.3 Radiation Exposure

The annual work and job function report is included as Appendix A and contains information pertaining to personnel radiation exposure. This information is included pursuant to Technical Specifications 5.6.1 and Regulatory Guide 1.16, Section C.1.b.(3).

The values are estimated doses for the listed activities based on direct reading dosimeter data. No correction factor was applied to the readings.

2.4 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 1997. 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 original 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 off-gas system activity and negative impacts on water chemistry parameters.

The Supply System formally notified the NRC of this change to the post-irradiation surveillance program.[2] The change was also incorporated into Amendment 50 (August 1995) of the WNP-2 FSAR.

1997 Results

Based on plant operational indicators, there was no evidence of failed fuel during Cycle 12. There was also no evidence of failed fuel for the calendar year 1997 portion of Cycle 13.

Inspections of ABB fuel assembly debris filters were performed during the R-12 Maintenance and Refueling Outage. The inspections were performed as a result of information from the fuel vendor that fretting wear had been noted on debris filters during post manufacturing out-of-reactor endurance flow tests. The reason for the failures was due to inadequate vendor testing and examination of the debris filters.

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2 Letter GO2-95-068, dated April 7, 1995, JV Parrish (SS) to NRC, "Modification of Post-Irradiation Fuel Surveillance Commitment"

As a result of the visual inspection effort, it was determined that there were broken springs in the debris filters and there were broken and loose spring turns in the bottom spacer span of some fuel assemblies. However, the inspection did not reveal any damage to the fuel assemblies, including the fuel rods.

A design change was implemented which consisted of removal of the filters, removal of the old lower fuel support pieces from the positions that contained the fuel assemblies with filters, and installation of a redesigned lower support piece. The lower support piece was installed under all of the affected fuel assemblies.

2.5 10CFR50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors"

This section contains information relative to changes and errors in Emergency Core Cooling System (ECCS) cooling performance models. This input is provided solely for informational purposes and ease of reference. There were no changes or errors since the last report. Regulation 10 CFR 50.46 only requires reporting when changes or errors have been identified.

Background

Included in this section is a description of the impact of any non-significant changes and errors discovered in the ECCS cooling performance evaluation models or in the application of such a model where the change or error was determined to be significant. For the purposes of this report, non-significant errors are those that are less than or equal to 50 degrees Fahrenheit. (Significant errors are reported pursuant to 10 CFR 50.72 and 10 CFR 50.73 and are not included in this report.)

Regulation 10 CFR 50.46 requires that, for each (non-significant) change to or error discovered in an acceptable ECCS cooling performance evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in 10 CFR 50.4.

Both General Electric (GE) and Asea Brown Boveri (ABB) methodologies are applied to the WNP-2 core. The GE methodology was used to license Siemens Power Corporation fuel. This is the LOCA analysis of record for WNP-2.[3] The ABB methodology was used to license ABB SVEA-96 fuel.[4]

1997 Results

There were no changes or errors in an ECCS LOCA analysis model or application since the last report, dated October 1997.

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3 General Electric Report NEDC-32115P, Revision 2, "Washington Public Power Supply System Nuclear Project 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," GE Nuclear Energy, July 1993

4 ABB Report CE NPSD-801-P, Revision 1, "WNP-2 LOCA Analysis Report," ABB Combustion Engineering Nuclear Operations, June 1996

2.6 10CFR50.59, "Changes, Tests and Experiments"

This section contains summaries of the Safety Evaluations (SE) completed for activities implemented during 1997 and is included pursuant to 10 CFR 50.59.

Regulation 10 CFR 50.59 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.

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.

2.6.1 Plant Modifications

The section contains information pertaining to implemented Plant Modification Records (PMRs) and is included pursuant to 10 CFR 50.59.

- **PMR 85-0528-00** (SE 94-142)
  (SE 95-104)

  This design change provided for the replacement and recalibration of overcurrent relays to allow for increased precision of inverse time-current settings and coordination with upstream and down stream overcurrent devices. These overcurrent relays provide overcurrent protection for the 4.16KV and 6.9KV buses and feeders and station service auxiliary transformers TR-S, TR-N1, TR-N2 and TR-B. This design change also provided for revised overcurrent relay and low voltage circuit breaker setpoints.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that this change would not impact the ability of the electrical distribution system to function and monitor overcurrent transients and accidents. The modification also does not change the functional design, logic or control scheme of any component or system. The proposed activity consists of resetting of existing relays and replacement of additional relays which are of the same electromechanical type in order to provide proper overcurrent coordination of the medium voltage switchgears.

  No new system failure modes have been added and no existing failure modes have been changed since the new relays are of the same electromechanical induction disk design. All other changes consist of recalibrations of existing equipment.

- **PMR 87-0244-06** (SE 97-075)

  This design change provided for several changes to the differential temperature logic for reactor recirculation flow control system cavitation protection. The differential temperature setpoint was increased from 9.9 degrees Fahrenheit to 10.7 degrees Fahrenheit. The differential temperature setpoint reset was increased from 10.9 degrees Fahrenheit to 11.2 degrees Fahrenheit. The differential time delay to runback was increased from 15 seconds to ten minutes. Alarm annunciation was changed from actual runback initiation to the start of the timing period. These changes were made following a review of the cavitation setpoint and methodology to determine if any runback value or cavitation logic changes were necessary due to problems that were encountered during testing of the reactor recirculation adjustable speed drive and digital feedwater level control systems.

The cavitation interlock protects the recirculation pumps and jet pumps from cavitation damage. The interlock signal is the differential temperature between the steam dome temperature (derived from steam dome pressure measurement) and the recirculation pump suction temperature. Increasing the pump speed above the minimum value is prevented if the temperature difference is less than the setpoint. Similarly, the pumps are automatically run back to the minimum speed if the setpoint is reached.

Safety Evaluation Summary

It was concluded from the safety evaluation that the cavitation interlock setpoint was being changed to reflect calculated values and includes margin and accounts for the high cavitation conditions which would be experienced under increased core flow. Increasing the existing time delay will result in the avoidance of reactor recirculation system pump speed runbacks. The alarm logic was being changed to initiate the alarm when the cavitation setpoint is exceeded, rather than the current design of after the time delay.

The recirculation flow control system does not perform any active safety function. The primary relation of the system to the licensing basis analysis is as an initiator of events. The proposed modification changes provide adequate equipment protection against cavitation damage.

The proposed changes provide for avoiding recirculation pump runbacks caused by false indication of cavitation conditions or by a setpoint that is inappropriate during some operating conditions. This will allow control room operators time to validate the alarm and determine if it is the result of an actual condition or the result of spurious component failures or other problems.

- PMR 90-0209-00 (SE 97-017)

This design change provided for installation of manual test valve assemblies on all eight main steam isolation valve accumulators. One of the existing plugs on each accumulator was replaced with a test valve. This will allow for venting of the pneumatic supplies to the valves for leak testing and provide for periodic blowdown of the accumulators to remove potential debris or moisture which may have migrated to the accumulators.

Safety Evaluation Summary

It was concluded from the safety evaluation that the pressure boundary integrity of the piping system and accumulator tanks is maintained. The passive component function of the accumulators will also be maintained with the addition of the new normally-closed test valve. The new test valve assemblies will function the same as the plug, which is to maintain accumulator tank pressure integrity.

The existing piping systems, structures and components were analyzed and remain qualified to the applicable ASME code and design requirements. The stresses associated with the new test valve assemblies were calculated and determined to be qualified and within system loading bases. During each outage, the accumulators and test valve assemblies will be checked for leak tightness.

- PMR 90-0279-00 (SE 96-076)

This design change provided for installation of freeze protection drain lines and valves for siphoning water from the down stream side of spray pond isolation valves SW-V-12A and SW-V-12B to facilitate draining above ground piping in the standby service water system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed modification was limited to installation of piping and manual valves in the standby service water system pump houses and the installation of a short extension of a vent line directly adjacent to Spray Pond A. The modification will have no physical or electrical interface with any other system.

The change will not impact operation of the standby service water system. These drain lines, in conjunction with operator actions, will ensure that the potential freezing of the system will not limit the capability of the accident mitigation functions of the standby service water system. The modification will also have no impact on spray pond water level, temperature or sedimentation Technical Specification requirements.

- PMR 91-0153-00 (SE 94-094)

This design change provided for installation of flushing connection tees on the floor and roof drain system piping risers in the diesel generator, turbine generator and radwaste buildings. These connections will provide a flow path for removal of water from the periodic flushing of fire protection sprinklers.

Safety Evaluation Summary

It was concluded from the safety evaluation that flushing of the fire protection system sprinklers is typically performed during plant outages. This modification does not permanently crosstie the fire protection system sprinklers with the drain system. The sprinkler system will be returned to its design configuration upon completion of flushing activities and, during normal operation, the drain flush connections will be capped.

This modification will not impact any fire protection sprinkler system design functions. The connections have been designed to the appropriate quality class and seismic requirements. Installation of the flushing connections and caps will also not impact the drain piping design function of removing water from the non-radioactive areas of the plant.

- PMR 91-0299-00 (SE 97-067)

This design change provided for the rerouting of piping to accommodate the use of a new corrosion and scale inhibitor used in treatment of the plant service water system. The modification resulted in the routing of two separate lines to system injection points. Heat tracing was also installed on the outside portion of the new chemical feed lines for freeze protection.

Safety Evaluation Summary

It was concluded from the safety evaluation that no plant systems and components required to mitigate the consequences of an accident previously evaluated will be impacted by this design change. The modification was performed on one plant service water system line at a time and post modification testing confirmed leak tightness. The plant service water system serves only non-essential systems and is not required to perform a safety function.

- PMR 92-0178-03 (SE 96-104)

This design change provided for the replacement of six Division 2 emergency diesel generator starting air pressure switches with low deadband components. The existing pressure switches had a large deadband, which impacted the ability of the components in meeting setpoint calculation tolerances. This change will allow for the switches to be reset within calculated permissible ranges. The change will also help ensure that air receiver pressure remains above the minimum required to provide five starts of the emergency diesel generator.

Safety Evaluation Summary

It was concluded from the safety evaluation that the pressure switch replacements would not adversely impact the safety function or performance of any plant system. The new switches are Quality Class 1 and perform a passive safety function.

The new switches have the same form, fit and function of the previous switches. The low deadband for the new pressure switches will allow the components to stop the air compressors prior to exceeding piping and air receiver design pressure limits. The pressure switches are fully qualified for a safety-related application.

- PMR 94-0043-01 (SE 96-083)

This design change provided for modification of the sealing mechanisms for reactor core isolation cooling system flex wedge valve RCIC-V-13 and residual heat removal system flex wedge valve RHR-V-42B to provide bonnet venting to the high pressure piping. The sealing mechanism was changed such that it seals on only one of the two wedge seats through the use of a bypass port. An external bypass line was added which equalizes the pressure between the wedge disc and the reactor side of the valve ends.

This action was a commitment made in response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves." The concern was of rapid reactor depressurization prior to the open signal for reactor core isolation cooling injection, and rapid reactor depressurization prior to the open signal for residual heat removal low pressure coolant injection.

Safety Evaluation Summary

It was concluded from the safety evaluation that the bypass line would not interfere with the performance of the valves or systems. This change enhances the reliability of the opening function of the valves through the elimination of pressure locking as a potential failure mode.

The change will provide a passive form of pressure equalization. The reactor isolation function is not affected. The non-reactor-side disc continues to perform the containment isolation function.

- PMR 94-0176-00 (SE 94-226)

This design change provided for an upgrade of the refueling platform control system. The existing relay controls for the bridge, trolley and main hoist were replaced with a programmable logic controller. The hydraulic load cells, dillon force switches and slack cable switch were replaced with strain gauge load cells, strain gauge transmitters and the programmable logic controller. The position display system, interlock display system and main hoist load display were replaced with a flat panel display. The main hoist motor, air system tubing and encoder positioning system were also replaced.

Safety Evaluation Summary

It was concluded from the safety evaluation that existing safety interlocks, hoist breaks and redundant hoist brakes for the main hoist, frame hoist and monorail hoist are maintained. A boundary zone function was added for the main hoist, which reduces hoist speed as the hoist

approaches obstructions such as fuel racks, fuel pool walls or the vessel wall. The boundary zone also restricts hoist movement in a predefined region around obstructions. This helps to prevent fuel or refueling mast damage in the unlikely event of collision with obstacles.

The only credible accident involving the refueling platform is a fuel handling accident where an object would be dropped into either the reactor vessel or the spent fuel pool. The platform is non-safety related, but Seismic Class 1. The new equipment was designed to meet Seismic 2/1 criteria. The consequences of a fuel handling accident are independent of the refueling platform and the refueling platform control system. A malfunction of the refueling platform control system places all hoists and motor drives in a fail-safe condition. In the case of loss of power, the refueling platform will be in a fail-safe condition (i.e., all brakes will engage and the hoist grapple will close.)

- **PMR 94-0334-00** (SE 96-086)

This design change provided for the removal of main steam system valve MS-V-5 and the associated motor operator from the reactor head vent system that is connected to Loop A of the main steam system in the drywell. There was a history of leakage through the valve packing and uncontrolled leakage through the packing had the potential to result in unacceptable drywell unidentified leakage. Previous efforts to correct the leakage were unsuccessful and the 250-lb valve was replaced with two-inch, Schedule 160, carbon steel piping.

Safety Evaluation Summary

It was concluded from the safety evaluation that MS-V-5 has no safety function either during or following a loss of coolant accident. The valve is normally open during reactor operation to remove noncondensable gases from the reactor vessel head space. The valve is a part of the reactor coolant pressure boundary and it performs a component function of pressure integrity to support the integrity of the pressure boundary. Replacing the valve with piping will ensure that the reactor head vent system will maintain the design functions of venting non-condensable gases, reactor pressure vessel hydrostatic venting and pressure boundary integrity.

The probability of occurrence of a previously evaluated accident is not increased by this change. Affected piping structures, systems and components were reanalyzed and are qualified to pertinent ASME code and WNP-2 design requirements. The change does not impact the safe shutdown analysis for pipe breaks inside containment. Reactor head vent system reliability is enhanced and maintained by this change.

- PMR 94-0364-02 (SE 96-011)

This design change provided for installation of a permanent shielding wall inside the drywell across the top of the control rod drive undervessel hatch and down both sides of the hatch for ALARA considerations. The shielding is supported from horizontal beams located between two vertical columns and the reactor vessel pedestal. The permanent structure eliminates the need to assemble tubelock scaffolding and install shielding every outage at this location.

Safety Evaluation Summary

It was concluded from the safety evaluation that installation of the shielding and the shielding support structure would not affect any system, structure or component that mitigates the consequences of an accident. The shielding and shielding support structure are qualified to appropriate design standards for seismic and hydrodynamic loads. The shielding support structure meets Seismic Category 1M requirements to prevent adverse 2/1 interactions with important to safety systems, structures and components.

The shielding 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 95-0134-00 (SE 95-063)

This design change provided for an addition of a logic trip channel to the logic design for traversing incore probe valve TIP-V-15. This change was made to ensure consistency with other Nuclear Steam Supply Shutoff System (NSSSS), Group 4, isolation valves. The existing logic is a single channel design. A minimum of two operable trip channels are required for NSSSS, Group 4, isolation valves when in Operational Modes 1, 2 or 3.

Safety Evaluation Summary

It was concluded from the safety evaluation that the addition of a second trip channel would not impact the accident mitigation function of the containment isolation system. The additional isolation logic channel will provide a redundant means of valve isolation in the event of a high drywell pressure or low reactor water level signal. Furthermore, the design and installation will comply with Quality Class 1 requirements.

The design also complies with the original design basis and functional requirements for containment isolation as described in the Final Safety Analysis Report.

- PMR 95-0269-00 (SE 97-063)

This design change provided for removal of the position indication and the "testable" feature of reactor core isolation cooling system check valve RCIC-V-66. This was a followup action to a previous temporary modification that removed the valve position indication lights from a control room panel. The position indicating shaft had become uncoupled from the valve hinges, rendering it inoperable. As part of this design change, the valve air actuator, air supply line and position switches were removed.

Safety Evaluation Summary

It was concluded from the safety evaluation that this modification does not change the safety function of the valve or the method of performing its safety function. The valve is a primary containment isolation valve and it will continue to perform its safety function as a check valve following removal of the actuator and position indication components. A shaft locking device was installed in place of the actuator to ensure that the disc-actuator shaft remains in a neutral (disengaged) position.

This change does not increase the probability of occurrence or consequences of accidents. In addition, the possibility of an accident or malfunction of equipment important to safety is not increased by this design change.

- PMR 96-0112-00 (SE 96-080)

This design change provided for the permanent installation of tanks and distribution piping associated with a hydrogen peroxide addition system for the standby service water system. The permanent installation replaces the temporary system that was installed to determine the effectiveness of controlling biological activity in the ultimate heat sink spray ponds and the standby service water system. The addition of hydrogen peroxide has shown to be very effective as a biocide and maintaining the heat transfer capability of the service water system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the peroxide addition system is not connected to any safety-related system and does not, as a system, affect the function of any safety-related components. Electrical power is not required for the gravity feed system. The two double-wall, polyethylene tanks are located north of Spray Pond B and will contain 3,000 gallons each of 50 percent hydrogen peroxide solution in water. All material associated with the installation are compatible with hydrogen peroxide and standby service water system components were evaluated and determined to be acceptable for the chemical treatment.

The only possible impacts on important to safety structures, systems and components relate to seismic effects, tornado effects, and the effects of the hydrogen peroxide on materials in contact with the system water. These potential impacts do not affect the ability of structures, systems or components to perform their intended safety function.

- PMR 96-0124-00 (SE 96-068)

This design change provided for the correction of electrical separation discrepancies in annunciator and computer direct "bridging" circuits. Appropriately sized fuses were installed downstream from the over-voltage contact point and upstream of the circuit entering the redundant raceway. This will result in isolation of potential fault currents in direct bridging non-1E circuits that may affect safety-related proximity circuits. Annunciation and computer inputs were also modified from a "close to alarm" status to an "open to alarm" status. This was necessary to provide an alarm condition when any of the newly-installed fuse functions are lost.

Safety Evaluation Summary

It was concluded from the safety evaluation that commitments to electrical separation criteria ensure that, during accident conditions with a postulated local fire, redundant safety functions cannot be affected. Making resulting non-1E circuit modifications in accordance with electrical separation criteria maintain the functionality of existing safety-related cable design parameters. Implementation of the alarm circuit changes, in accordance with single failure criteria, will have no impact on any accident or transient analysis.

There is no change to the operation of any plant system. The change in component operation (open to alarm) is fail-safe and similar to other components or circuits within the system. Adherence to single failure criteria and the use of alternative instrumentation for out-of-service alarms and computer points will not reduce the margin of safety as defined in the basis for any Technical Specification.

- PMR 96-0133-00 (SE 97-005)

This design change provided for installation of an orifice on the spray bypass discharge to the spray ponds in the standby service water system. Installation of the orifice on the discharge of the bypass line will reduce flow in the spray bypass mode to approximately the same flow as the spray mode and will provide backpressure in system piping to mitigate cavitation.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification would not alter the function of the standby service water system. The orifice serves no safety function. Cavitation downstream of service water flow elements SW-FE-1A and SW-FE-1B has been slowly progressing over a ten-year period. Inspections of the piping downstream of the flow elements will be conducted at five-year intervals to assure continuous knowledge of the condition of the piping and assess the effectiveness of the design change.

Post modification testing was required to ensure that the design flow rate is maintained in the spray bypass mode of operation. The potential effects on standby service water system heat removal capability and water inventory, as well as potential failure modes and associated impacts, were considered. It was concluded that the system safety function would not be compromised by this design change. In addition, Technical Specification limits on spray pond water level, water temperature and sedimentation accumulation are maintained.

- PMR 96-0140-00 (SE 97-006)

This design change provided for replacement of existing process sampling and reactor closed cooling system valves with valves of a new design to correct leakage problems. These valves are containment isolation valves and are used to sample reactor water at rated pressure. The new valves are supplied with flanges and electrical connectors to facilitate maintenance activities.

Safety Evaluation Summary

It was concluded from the safety evaluation that the valves will meet Quality Class 1 and ASME code requirements. The flanges comply with ASME code requirements. The change is limited to replacement of existing solenoid sampling valves with a new model by the same manufacturer. The replacement valves differ in that they were installed with flanges and have an electrical connector instead of field cables terminated at terminal strips in the valve cover. The electric connectors are qualified.

The new valves have a higher seating force and, as a result, are less likely to leak. Operation of the sampling valves is unchanged and post-maintenance testing verified operability of the new valves and the ability to meet closing time and local leak rate testing requirements.

- **PMR 96-0159-00** (SE 97-015)

This design change provided for installation of flushing tees and spectacle flanges to allow for high pressure/high flow flushing of floor drain system piping. Debris collects in the low point piping between the drywell floor drain sump and primary containment isolation valves FDR-V-3 and FDR-V-4. Typical flows through this floor drain piping are less than five gpm and are insufficient to completely flush the material out of the low point and into the sump in the high pressure core spray pump room. There is the potential that gradual deposition of the debris material between the ball and seat assemblies of the floor drain valves may result in longer close stroke times. This modification allows the floor drain system loop seal piping located in the wetwell to be flushed free of debris.

Safety Evaluation Summary

It was concluded from the safety evaluation that operation of the floor drain system will not be impacted by this design change. The flushing tees are blanked off during normal plant operation and will only be used for flushing during outages. The spectacle flanges will normally be in the open position during plant operation and are blanked off during flushing activities.

The proposed modification does not have any physical or electrical interfaces with any other system than the floor drain system. In addition, the components have been designed to the appropriate quality class and seismic requirements.

- **PMR 97-0014-00** (SE 97-008)

This design change provided for revision of the software for reactor recirculation system adjustable speed drive channels A1 and A2. New channel trips were added to supplement the existing time-delayed fault trip and alarm functions which will provide an automatic trip system in the event certain faults occur within a specified time period. Specifically, the time-delayed fault trips, source instantaneous overcurrent trip, load overcurrent trip, source phase locked loop error, load phase locked loop error and the gate turn off freeze alarm inputs will be monitored for events of unit time (in any combination) and provide a channel trip when predetermined limits are exceeded.

Safety Evaluation Summary

It was concluded from the safety evaluation that the software modifications reduced the need for operator intervention in the event of adjustable speed drive control faults. The potential for reactor recirculation system transients is mitigated because the control logic will automatically

monitor fault trips. This will provide an automatic trip system in the event that certain faults occur within a specified time, rather that relying on time-delayed and sustained signals or operator intervention due to annunciation signals.

The trips and diagnostic monitoring features do not affect the operation of the reactor recirculation system. No new accidents or transients are created, previously analyzed accidents and transients remain unaffected, no new malfunctions beyond those previously evaluated are created, and the Technical Specification margins of safety are maintained with this modification.

- PMR 97-0014-01 (SE 97-047)

This design change provided for revision of the software for reactor recirculation system adjustable speed drive channels B1 and B2. New channel trips were added to supplement the existing time-delayed fault trip and alarm functions which will provide an automatic trip system in the event certain faults occur within a specified time period. Specifically, the time-delayed fault trips, source instantaneous overcurrent trip, load overcurrent trip, source phase locked loop error, load phase locked loop error and the gate turn off freeze alarm inputs will be monitored for events of unit time (in any combination) and provide a channel trip when predetermined limits are exceeded.

Safety Evaluation Summary

It was concluded from the safety evaluation that the software modifications reduced the need for operator intervention in the event of adjustable speed drive control faults.. The potential for reactor recirculation system transients is mitigated because the control logic will automatically monitor fault trips. This will provide an automatic trip system in the event that certain faults occur within a specified time, rather that relying on time-delayed and sustained signals or operator intervention due to annunciation signals.

The trips and diagnostic monitoring features do not affect the operation of the reactor recirculation system. No new accidents or transients are created, previously analyzed accidents and transients remain unaffected, no new malfunctions beyond those previously evaluated are created, and the Technical Specification margins of safety are maintained with this modification.

- PMR 97-0040-00 (SE 97-038-00)

This design change provided for implementation of the WNP-2, Cycle 13 core design. The proposed activity consisted of operation of the Cycle 13 reload core with core thermal limits which have been developed with NRC-approved methodologies. The thermal limits are

specified in the Core Operating Limits Report (COLR). The change also included modification of in-core, Cycle 12, SVEA-96 fuel assemblies to remove debris filters. This also includes installation of special tools to remove the debris filters.

Safety Evaluation Summary

It was concluded from the safety evaluation that operation of Cycle 13 within the thermal limits defined in the COLR does not increase the consequences of the analyzed anticipated operational occurrences or accidents. 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.

The modification to install special tools and remove the debris filters meets fuel design criteria. The special tools used to remove the filters meet seismic and fuel pool heavy load requirements. Testing of the core is performed with plant procedures which have been in use for several cycles. Testing associated with debris filter removal effort simply involves dimensional measurement and inspection.

- PMR 97-0040-00 (SE 97-038-01)

This revised design change provided for an increase in the reactor recirculation system flow runback speed for the loss of feedwater pump trip event. The reactor recirculation system pump runback setpoint was changed from 27 Hz to 30 Hz based on the new power-to-flow map boundaries that were established with the Cycle 13 core design. This change was made as part of a follow-up review to determine if any runback value changes were necessary due to problems that were encountered during testing of the reactor recirculation adjustable speed drive and digital feedwater level control systems.

Safety Evaluation Summary

It was concluded from the safety evaluation that the design function of the reactor recirculation adjustable speed drive runback setpoint is to reduce recirculation system demand to within the capacity of one reactor feedwater system pump following the trip of a second pump. Flow runback events can lead to entry into the stability exclusion region or reactor power oscillations. Changing the adjustable speed drive runback setpoint to 30 Hz will reduce the risk of entry into the stability exclusion region of the power-to-flow map.

This change will also result in increased core flow after a runback than the original 27 Hz setpoint. Increased core flow benefits reactor stability. The 30 Hz setpoint is within the capacity of the reactor feedwater system. Therefore, the risk of a Level-3 or Level-8 reactor trip is not increased by the change. This change improves plant reliability by reducing the risk of unnecessary transients or a stability-related reactor scram.

- PMR 97-0068-00 (SE 97-074)

This design change provided for the re-routing of equipment drains in the main steam tunnel to the reactor building floor drain system to compensate for a plugged steam tunnel floor drain. The drain was blocked with grout during plant construction and efforts to open the drain have been unsuccessful. Floor drainage from the main steam tunnel was routed to radioactive floor drain sump FDR-SUMP-R4 in the residual heat removal system "C" pump room.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change would result in the introduction of a small amount of high purity wastes to the floor drain system. There is no impact on sump leak detection capability or the associated control room leak detection alarm function. Flooding considerations from the main steam tunnel are bounded by other postulated pipe breaks in the reactor building. The re-routing of the main steam tunnel drains to the floor drain system does not affect the probability or consequences of previously evaluated accidents and transients.

- PMR 97-0085-00 (SE 97-069)

This design change provided for installation of restrainer wedges on reactor recirculation system jet pumps MS-JP-14 and MS-JP-18. The restrainer wedges will reestablish the three-point contact between the inlet mixer and the restrainer bracket for each of these jet pumps to resolve an issue of gaps discovered between the set screws of the restrainer brackets and the inlet mixers for the pumps. This change provides the lateral restraint function originally provided by the jet pump inlet mixer setscrews.

Safety Evaluation Summary

It was concluded from the safety evaluation that the jet pump design includes establishing three contact points between the jet pump inlet mixer and the jet pump restrainer bracket. This design change restores the three-point contact design.

The safety function of the jet pumps is to maintain core flooding to two-thirds of the height of the core in the event of a recirculation system loss of coolant accident. Installation of the restrainer wedges will restore the degree of lateral support required for the original design configuration. This design change does not result in any changes in performance of the design features or the safety function for the jet pumps.

2.6.2 Temporary Modifications and Instrument Setpoint Changes

This section contains information pertaining to implemented Temporary Modification Requests (TMRs) and is included pursuant to 10 CFI 50.59.

TMR 97-015 (SE 97-066)

This temporary modification provided for temporary power to Division 2, 24 VDC, battery chargers during an SM-8 bus outage. The normal feed source was changed to a non-safety related local convenience outlet. This modification was necessary to ensure continued battery charging and that the source range monitors remained operable in accordance with the Technical Specifications to support fuel movement.

Safety Evaluation Summary

It was concluded from the safety evaluation that, during the operational mode in which this temporary modification was implemented (refueling), there were no applicable accident or transient analyses that require a safety-related power source to the charger. The battery provides power to the source range monitors which are required to be in operation during the refueling mode.

The non-safety related power source will provide the necessary power to keep the battery charged and, therefore, the source range monitors would remain functional and operable. In the event that the alternate power source became unavailable, the battery would continue to provide power to the source range monitors until fuel movement was suspended and the monitors declared inoperable. No safety functions, accident consequences, equipment malfunctions or margins of safety are affected by this temporary modification.

2.6.3 FSAR Changes

This section contains information pertaining to FSAR Licensing Document Change Notices (LDCNS) and Change Notices (SCNs) and is included pursuant to 10 CFR 50.59.

- **LDCN-FSAR-96-035** (SE 96-056)

  This LDCN provided for a change to the FSAR to delete the generic requirement for fire breaks for vertical cable trays with continuous runs of 30 feet or more. In addition, the FSAR was revised to reflect that coated fire breaks have been abandoned in place and that existing silicone foam penetration seals are maintained as credited fire breaks. These changes were the result of resolution of issues pertaining to the use of Thermo-Lag and Flamemastic material as coating for cable trays.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that fires are evaluated design basis events that can affect important to safety structures, systems and components. However, abandoning the coated fire breaks does not impact plant systems and components required to mitigate the consequence of accidents.

  Furthermore, fire breaks are neither essential nor needed to ensure post-fire safe shutdown. Vertical cable trays without fire breaks, or with inoperable fire breaks, have no impact on credited fire barriers, or fire suppression or fire detection systems which protect plant equipment important to safety. In the event of a fire, defense-in-depth design features are also available to ensure detection and suppression prior to significant fire propagation. These features include qualified cables, low cable fill, raceway enclosures, lack of adjacent hazards, lack of adjacent ignition sources, area detection systems and area suppression systems.

- **LDCN-FSAR-96-044** (SE 97-014)

  This LDCN provided for a revision to the FSAR to reflect a change in the maintenance frequency from 18 months to 24 months for inspection of the emergency diesel generators. This was initiated in response to a planned change to the WNP-2 maintenance cycle from 12 months to 24 months.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that extending the inspection frequency from 18 to 24 months would still be within the vendor-recommended frequency. The vendor concurred that a 24-month frequency is acceptable, considering the mild environment and type of service associated with the emergency diesel generators.

This change was also evaluated in accordance with the guidance provided in Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators." In Generic Letter 94-01, the NRC advised all licensees that extending the 18-month surveillance interval to 24 months could be justified on the basis of a successful equipment performance history. Reviews of historical maintenance and surveillance data have shown that the tests normally pass the surveillances at the current frequencies. An evaluation was performed using this data and it was concluded that extending the frequency would have no impact on the safety function of systems or components.

- LDCN-FSAR-96-052 (SE 96-071)

This LDCN provided for a revision to the FSAR to correct errors and inaccuracies that were discovered during a review of all references to the containment instrument air system. The changes included: Removal of reference to the number of automatic depressurization system safety relief valve cycles available from the nitrogen bottles; correction of reference to location of nitrogen bottles on line; removal of reference to monitoring of the local indication of the bottles on line; removal of references to containment instrument air system compressors; correction of references to compressors in the control and service air systems; removal of the statement that the containment instrument air system is manually initiated by the system start control switch; and the addition of a description of the automatic function in support of the automatic depressurization system.

Safety Evaluation Summary

It was concluded from the safety evaluation that removing reference to the number of automatic depressurization system valve cycles available from the nitrogen bottles would have no affect on the function of the containment instrument air system or the capability of the system to meet its design requirements. Removal of reference to remote indication of the number of nitrogen bottles on line outside the reactor building does not impact the function of the system. The remote indication is not relied upon for any of the accident mitigation functions performed by the system. There is sufficient control room indication to determine the status of the system.

The air compressors in the containment instrument air system were evaluated and physically removed from the plant during implementation of PMR 90-0026. Removal of those remaining FSAR references that were missed during that design change and associated FSAR update has no impact on the function of the system and is consistent with the 10 CFR 50.59 safety evaluation that was performed in support of PMR 90-0026. It was concluded from the PMR 90-0026 safety evaluation that removal of the air compressors in the containment instrument air system did not increase the possibility of occurrence of an accident, or malfunction of equipment important to safety. The air compressors were not safety-related and were not being used or credited in the accident analyses.

The correction pertaining to references to the compressors in the control and service air systems was made to align the references with the current plant configuration and FSAR system descriptions.

Removal of the reference to manual initiation of the containment instrument air system does not impact the design basis function of the system. The system is normally in service whenever the automatic depressurization system is required to be operable. The backup nitrogen bottles are automatically activated when the normal source cannot maintain adequate system pressure. Furthermore, manual initiation capability is part of the design of the automatic depressurization system and the system supply headers are normally pressurized.

- **LDCN-FSAR-96-055** (SE 96-094)

This LDCN provided for a revision to the FSAR to delete reference to disposal of radioactive material by means of the sanitary sewer system. This change corrects the FSAR to reflect the new requirements of 10 CFR 20. This regulation no longer allows disposal of radioactive material through the sanitary sewer system.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change ensures compliance with 10 CFR 20 requirements pertaining to disposal of radioactive waste. This change does not affect any system, structure or component required for safety or important to safety operation. The change simply reflects implementation of regulatory change which conservatively expands the control of radioactive effluents.

- **LDCN-FSAR-96-060** (SE 96-070)

This LDCN provided for a revision to the FSAR to reflect the use of potable water as an additional source of makeup water for the standby service water system spray ponds (ultimate heat sink). A makeup water line was installed to Spray Pond B from a connection to the potable water system at the supply line to the waste treatment facility. The normal makeup source is from the tower makeup system (river water). During the summer, considerable biological life is introduced into the ponds from the river water. This results in the use of relatively large quantities of hydrogen peroxide, the primary water treatment chemical. The use of potable water makeup is expected to result in a more stable water chemistry in the standby service water system.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change would not impact any safety-related systems. The only potential impact on important to safety systems or components is the possibility for some flooding in the event of a potable water makeup line failure. The new makeup line is located in an area where topography ensures that water released from a line failure would flow away from the plant and the spray pond area. The potential failure of the new line does not pose a threat to the safety function of any plant system or the ultimate heat sink. A makeup line failure would be identified by routine surveillance of the site perimeter by the security staff, using both video cameras and patrols.

The design also does not have the potential to cause ultimate heat sink water inventory loss, such as siphoning. Specific limitations on standby service water chemistry, such as chloride concentration, are met by the potable water.

- LDCN-FSAR-96-074 (SE 96-088)

This LDCN provided for a revision to the FSAR to reflect deactivation of the electronic overspeed trip function on the reactor core isolation cooling system turbine. The electronic overspeed trip had been a source of spurious trips that have had an adverse effect on the reliability of the reactor core isolation cooling system. Deactivation of the electronic overspeed trip was recommended by General Electric, the system designer, in Service Information Letter 382, "Removal of RCIC Electronic Overspeed Trip."

Safety Evaluation Summary

It was concluded from the safety evaluation that there would be no adverse impact on the system as a result of the modification. System reliability is improved by elimination of spurious trips attributed to the electronic trip circuit an associated contactor. Turbine protection from overspeed is still maintained by the mechanical trip feature.

Other electrical trips such as high turbine exhaust pressure, low pump suction pressure and the isolation signal are not affected by this modification. Deactivation of the electrical overspeed trip from the turbine control logic has no impact on operation of the reactor core isolation cooling system, except to increase system reliability.

- LDCN-FSAR-96-075 (SE 96-089)

This LDCN provided for a revision to the FSAR to reflect a change to a WNP-2 engineering design specification that would allow the use of an acceptable equivalent A/SA grade material for installed fasteners used in the control rod drive system, the main steam safety/relief valves and the steam dryers and separators.

Safety Evaluation Summary

It was concluded from the safety evaluation that the use of A/SA 194 Grade 7 material (alloy steel) as an alternate for Grade 2H material (carbon steel) was acceptable. The new material has the same hardness range as the original, indicating similar overall strength properties. The Grade 7 material has a high fracture toughness and superior corrosion resistance which results in greater durability and performance.

It was also determined that the use of A/SA 194 Grade 8A material (stainless steel) as an alternate for Grade 8 material (stainless steel) was also acceptable. The two materials are identical in property and chemistry, with the exception that the Grade 8A material has improved corrosion resistance due to additional manufacturing process steps. Improved corrosion resistance will improve fastener life.

No new failure modes have been introduced and the overall function of the fasteners has not changed.

- **LDCN-FSAR-96-078** (SE 96-108)

This LDCN provided for a revision to the FSAR to reflect an increase in the maximum hydrogen operating pressure of the main generator from 75 psi to 78 psi. This change, with vendor (Westinghouse) concurrence, does not increase the output capability of the generator but will allow operators additional time between required hydrogen additions when load demands require operation near the capability curve. Operation near the capability curve requires more frequent additions to the hydrogen system.

Safety Evaluation Summary

It was concluded from the safety evaluation that implementation of this change does not impact the pressure rating of the hydrogen system. The increased pressure is below the design pressure rating of the system and within Westinghouse-recommended hydrogen pressure alarm settings. The change does not alter the function or decrease the availability of the hydrogen system.

- **LDCN FSAR-96-082** (SE 96-095)

This LDCN provided for a revision to the FSAR to reflect that the rod sequence control system is redundant to the rod worth minimizer. In addition, the change reflected the lowering of the rod worth minimizer low power setpoint from 20 percent to 10 percent rated thermal power.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change does not increase the probability or consequences of previously evaluated transients or accidents. The change has no impact on the banked position withdrawal sequence. This sequence reduces the consequences of a continuous rod withdrawal during reactor startup and a control rod drop accident. Making the rod sequence control system redundant to the rod worth minimizer and reducing the low power setpoint does not reduce the margin of safety as defined in the bases of any Technical Specification.

The rod worth minimizer and rod sequence control systems are separate systems that are neither required for or support operation of other systems. When the rod worth minimizer system is operable, the rod sequence control system is not needed because the rod worth minimizer prevents deviation from prescribed banked position withdrawal sequence patterns. In the event the rod worth minimizer is out of service, control rod movement and compliance with the sequence pattern is required to be verified by a second licensed operator or qualified member of the technical staff.

- LDCN-FSAR-96-093 (SE 97-001)

This LDCN provided for a revision to the FSAR to reflect changes in the insulation classification allowed for high pressure core spray, low pressure core spray, residual heat removal and service water system pump motors. The insulation class was changed from "B" to "B, F or H." This change will allow for flexibility in the procurement of replacement motors.

Safety Evaluation Summary

It was concluded from the safety evaluation that allowing the use of other insulation classifications would not affect the design safety functions of the systems involved. The proposed change would also not impact the reliability or environmental qualification of the pump motors that are replaced with comparable motors that have class "F" or "H" insulation.

The insulation temperature ratings are greater than the sum of the motor temperature rise, the ambient temperature at the motor location, and the hot spot temperature allowance. Any future motor replacements will be subject to the same requirements for equipment qualification, regardless of insulation class.

- LDCN-FSAR-96-094 (SE 96-098)

This LDCN provided for a revision to the FSAR to reflect that the refueling floor jib cranes can be used to transfer new fuel from the new fuel storage vault to the fuel storage pool. The FSAR was also revised to reflect proper methods of performing non-destructive examinations of the reactor building crane hooks and to remove the requirements for chaining hoists inside containment when they are not in use.

Safety Evaluation Summary

It was concluded from the safety evaluation that the reactor building crane hooks, the refueling floor jib cranes and the hoists inside of containment do not mitigate the consequences of any design basis accident, transient or malfunction. The only mechanism for increasing the probability of an accident, transient or malfunction affected by this change is a structural failure of reactor building crane hooks, the refueling floor jib cranes and the hoists inside of containment. However, a structural failure of these components is not increased by these changes.

Appropriate non-destructive examination of the reactor building crane hooks will be performed to ensure structural reliability. The refueling floor jib cranes are qualified to transfer new fuel from the new fuel storage vault to the spent fuel pool. The hoists inside containment have been qualified for dynamic loads and no longer need to be chained. Furthermore, this change does not affect the interlocks and physical stops that prevent crane travel over the spent fuel pool.

- LDCN-FSAR-96-098 (SE 96-111)

This LDCN provided for a revision to the FSAR description of design requirements pertaining to excess flow check valves. The FSAR was revised to clarify that excess flow check valves do not perform an active containment isolation function for any analyzed accident requiring containment isolation. There were no physical changes made to the facility as a result of this clarification.

Safety Evaluation Summary

It was concluded from the safety evaluation that excess flow check valves provide a passive boundary function and perform no active safety function. There are no credited valve closure design features assumed in the accident analyses for mitigation of an instrument line break.

The instrument lines are Seismic Category 1 and are assumed to maintain integrity for all accidents, except for the instrument line break accident. Closure of excess flow check valves is not credited for mitigating an instrument line break. There is no impact, either increase or decrease, to onsite or offsite dose consequences as a result of this clarification.

- LDCN-FSAR-96-099 (SE 97-004)

This LDCN provided for a revision to the FSAR to reflect the actual configuration of the main steam system following a detailed system walkdown and review of system design requirements by Engineering personnel. The FSAR was revised to reflect that the maximum pressure differential across the main steam line flow restrictor is 10 psi at 100 percent rated flow instead of 6.3 psi. The FSAR was also revised to reflect the correct isolation logic configuration for several valves.

Safety Evaluation Summary

It was concluded from the safety evaluation that the maximum pressure differential of 10 psi across the main steam line flow restrictor is bounded by the design condition to limit steam flow in a severed line to less than 200 percent of rated flow.

The inboard and outboard steam line drain valves are powered from separate divisional power and are installed in accordance with design specifications. No physical changes were required by this FSAR revision. The change simply corrects errors in the description of the main steam system.

- LDCN-FSAR-97-001 (SE 97-002)

This LDCN provided for a revision to the FSAR to reflect the replacement of residual heat removal system water leg pump motor RHR-M-P/3. The new motor is identical to and has the same horsepower as the original motor, but draws slightly fewer amps because of a more efficient design.

Safety Evaluation Summary

It was concluded from the safety evaluation that the function of the residual heat removal system water leg pump motor was not affected by this change. The motor replacement is electrically acceptable and will have no impact on the electrical distribution system or the associated loads.

The replacement of the motor also will not affect the ability of the reactor protection system in response to a reactor water level or pressure transient.

- **LDCN-FSAR-97-008** (SE 97-050)

This LDCN provided for a revision to the FSAR to reflect changes in service water system flow rates and heat loads used in the ultimate heat sink analysis. The changes were made based on actual plant operating data and current system alignments. For example, flow rate to the residual heat removal system "A" and "B" seal coolers has decreased, the residual heat removal system "C" cooler no longer uses service water coolant, the lineup of service water to the control room chiller has been substituted for radwaste mixed air-supplied cooling coils, and the hydrogen recombiner scrubber flow has been removed since this flow rate does not return to the spray ponds. The net result is a change in Division 1 total flow through the system from 10325 gpm to 10351 gpm. For Division 2, total flow through the system dropped to 10316 gpm. For Division 3, total flow through the system dropped to 952 gpm.

Safety Evaluation Summary

It was concluded from the safety evaluation that none of these changes adversely impacts safety. The addition of the higher heat load and flow rate pertaining to the change from the radwaste mixed air-supplied cooling coils to the chiller has already been incorporated into the ultimate heat sink analysis. The additional load and flowrate are already reflected in the results listed in the FSAR.

Because of the small magnitude of the change, the removal of the hydrogen recombiner scrubber flowrate and heat load and reduction in residual heat removal system "A" seal cooler flow will have no positive or negative impact on spray pond temperatures or inventory. The reduction in high pressure core spray service water system flow rate has no impact because it is no longer used in the ultimate heat sink analysis. The changes in the service water "B" loop will have no impact because Division II flow rates and heat loads are not used in the ultimate heat sink analysis.

- **LDCN-FSAR-97-014** (SE 97-036)

This LDCN provided for a revision to the FSAR to reflect a change to the requirement that implementing procedures for the release of systems and equipment for maintenance or surveillance testing and for return to service may be delegated to a senior reactor operator in the control room. The requirement was modified to allow for work authorization authority to be delegated to an on-shift licensed senior reactor operator.

Safety Evaluation Summary

It was concluded from the safety evaluation that the basis for allowing delegation to an on-shift senior reactor operator is that the activity will be performed with provisions to ensure that the shift manager is kept fully informed of system status. The release of systems and equipment continue to be performed by a senior reactor operator. The shift manager will continue to be fully informed of system status through daily plant meetings, shift turnovers, the daily plant schedule and the work management system.

Operator actions assumed in the FSAR remain unchanged. Any release of systems for maintenance or surveillance must be accomplished within licensing basis restrictions, even if the action is being taken by a senior reactor operator located outside of the main control room.

- LDCN-FSAR-97-015 (SE 97-016)

This LDCN provided for a revision to the FSAR to reflect an agreement that was reached with the U.S. Department of Energy for the treatment of sanitary wastes from the Hanford 400 Area in the Supply System sanitary waste treatment facility. The Hanford 400 Area consists primarily of the Fast Flux Test Facility.

Safety Evaluation Summary

It was concluded from the safety evaluation that the treatment of Hanford 400 Area sanitary waste in the Supply System sanitary waste treatment facility has no impact on the safe operation of WNP-2. The sanitary waste treatment facility is neither an initiator of, nor contributor to, accidents or transients. The facility has no impact on either safety-related or important to safety structures, systems or components that prevent or mitigate accidents. The possible introduction of slightly contaminated sewage into the sanitary waste treatment facility has no safety significance pertaining to personnel or the operation of WNP-2.

Although the Hanford 400 Area wastes may introduce low concentrations of radioactive materials (principally tritium) into the sanitary waste treatment facility, concentrations will be such that there is no concern from the perspective of human health and safety.

- LDCN-FSAR-97-019 (SE 97-045)

This LDCN provided for a revision to the FSAR to correct a table notation pertaining to plant switchgear and motor control center power supplies. The table was revised to show that DC control power panel DP-S1-1F supplies power to several switchgears. Panel DP-S1-1D was incorrectly shown as being the power source. Although the switchgears were originally supplied control power by DP-S1-1D, control power was changed to DP-S1-1F by means of a design change in 1983. However, the FSAR was not updated to reflect the change.

Safety Evaluation Summary

It was concluded from the safety evaluation that there was no plant safety impact as a result of switching breaker control to a different DC power supply. Control power panels DP-S1-1D and DP-S1-1F are both in the same division, are equally as reliable as a safety-related power supply and have the same power source.

Both panels are qualified for the same conditions of service and are completely interchangeable in fit, form and function.

- LDCN-FSAR-97-021 (SE 97-035)

This LDCN provided for a revision to the FSAR to reflect the deletion of the requirement that fire suppression be made available for inoperable essential Halon systems.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change does not impact the ability to achieve and maintain safe shutdown. The only WNP-2 essential Halon systems are located in the continuously-staffed main control room. Existing manual fire suppression systems available to the control room staff are adequate without the need for staging additional fire equipment. Eight portable fire extinguishers (one dry chemical and seven Halon units) are staged in the main control room. Fire hose stations are also located in close proximity to three of the four main control room entry and egress points.

Furthermore, the WNP-2 Safe Shutdown Analysis assumes all components in the main control room are destroyed in the event of a design basis fire and also accounts for spurious operation of associated circuits. The previously-evaluated impact on components already assumes the worst case scenario of all equipment lost within a fire area. In addition, the 10 CFR 50, Appendix R, post-fire safe shutdown analysis does not credit the operability of the power generation control complex Halon system

- LDCN-FSAR-97-022 (SE 97-037)

This LDCN provided for a revision to the FSAR to reflect a change in the minimum staffing requirement for health physics technicians from two to one.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change in minimum staffing for health physics technicians from two to one does not impact previously-analyzed accidents or transients. The probability or consequences of accidents analyzed in the FSAR is unaffected by health physics technicians.

Health physics technicians are not credited in any way with initiating, mitigating or preventing accidents analyzed in the safety analysis. The change in the minimum number of health physics technicians is consistent with the Improved Technical Specifications which contain the statement that, "An individual qualified to implement radiation protection procedures shall be on site when fuel is in the reactor." The change is also consistent with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," and industry practice.

- LDCN-FSAR-97-024 (SE 97-030)

This LDCN provided for a revision to the FSAR to reflect that post-accident sampling system refresher training, consisting of classroom instruction and a practical exercise, will be provided annually. This is a change from the current requirement to provide refresher training every six months.

Safety Evaluation Summary

It was concluded from the safety evaluation that the design, construction and operation of the post-accident sampling system was not changed. Procedures controlling the operation of the system were also not changed. Furthermore, this change has no impact on the system capability for sampling reactor coolant, plant gaseous effluents and containment atmosphere following an accident.

Annual training will maintain the capability of personnel assigned to obtain a post-accident sample in the time required.

- LDCN-FSAR-97-025 (SE 97-086)

This LDCN provided for a revision to the FSAR to reflect a revision to the emergency diesel generator loading tables and clarify the text description pertaining to the loading tables.

Safety Evaluation Summary

It was concluded from the safety evaluation that the safety function of the diesel generators was not impacted by this change and no other equipment is affected. The updated values presented in the tables are based upon a diesel generator loading calculation which incorporates the cumulative results of all plant-implemented design changes.

The increase in loading is within the 2,000 hour/year, 8,000 hour/year, 30 minute/year and brake horsepower ratings specified for each diesel generator, except for the loading for DG-3. The automatically-connected, steady-state loading for DG-3 continues to slightly exceed the 8,000 hour/year rating of 2600 kW. However, the steady state loading remains well below the 2,000 hour/year rating specified in the FSAR.

The diesel generators remain within design limits as a result of this change and the change has no impact on safety.

- LDCN-FSAR-97-028 (SE 97-042)

This LDCN provided for a revision to the FSAR to reflect an enhanced description of several activities pertaining to reactor pressure vessel disassembly and reassembly.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change does not impact the fuel handling, inadvertent control rod withdrawal, misplaced bundle, control rod drop or failure of residual heat removal system shutdown cooling accidents. The change simply reflects clarification of the FSAR detail pertaining to the reactor pressure vessel disassembly and reassembly work methods.

These details are not relied upon to mitigate the consequences of equipment malfunction. The details were revised to reflect industry-wide methods for reactor pressure vessel disassembly and reassembly.

- LDCN-FSAR-97-029 (SE 97-046)

This LDCN provided for a revision to the FSAR to correct editorial errors and make minor changes to the organizational structure and references, and to expand the descriptions of waste processing controls for clarification and alignment with current practices and processes pertaining to the solid waste management system.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change would have no impact on either safety-related or important to safety structures, systems or components. The solid waste and related systems do not interact with, and are physically separated from, any structures, systems or components credited in accident or transient analyses.

The changes are primarily editorial or clarifying in scope and were made to update the FSAR to reflect current practices pertaining to solid waste management.

- **LDCN-FSAR-97-030** (SE 97-071)

This LDCN provided for a revision to the FSAR to reflect an update to the design and operating temperature and pressure values for the reactor closed cooling system and to remove references to the decontamination concentrators which have been deactivated. The design temperature and pressure values remain within stated design limits.

Safety Evaluation Summary

It was concluded from the safety evaluation that there were no physical changes to the plant as a result of this change. The only safety-related function of the reactor closed cooling system is the preservation of containment integrity by means of the containment isolation valves on the supply and return lines. This proposed activity has no impact on the containment isolation function.

Therefore, the ability of the system to meet its safety function is maintained. The functional capability of the reactor closed cooling system remains the same before and after the change.

- **LDCN-FSAR-97-031** (SE 97-053)

This LDCN provided for a revision to the FSAR to reflect deletion of reference to the freon degreaser unit in the plant decontamination facility system description. The freon degreaser was removed during 1995 and replaced with a plastic media blaster to eliminate waste concerns. However, the FSAR was not updated at the time to reflect the change.

Safety Evaluation Summary

It was concluded from the safety evaluation that the FSAR description of the decontamination facility is sufficiently detailed without reference to either of these components. The freon degreaser, DCN-FDG-1, was a non-safety related, Quality Class 2, Seismic Class 2,

component located in a non-safety related area of the plant (decontamination facility). This equipment was used to decontaminate small items or components. The degreaser had no important to safety function.

The plastic media blaster will perform the same function in the same area of the plant. Neither the removal of the freon degreaser from the plant nor the installation of the plastic media blaster in the plant will affect any important to safety structure, system or component.

- LDCN-FSAR-97-035 (SE 97-043)

This LDCN provided for a revision to the FSAR to reflect an update of the battery/system load tables for the Division 1, 2, and 3 safety-related 125 VDC and 250 VDC batteries. The battery duty cycles described in the tables were changed based on revisions to battery sizing calculations. Duty cycles provide the basis for test values used in the battery surveillance procedures.

Safety Evaluation Summary

It was concluded from the safety evaluation that these changes were minor in nature and the existing surveillance procedures bound the revised values. Although constituent loads varied, the bottom line total amperes defined in the duty cycles remained essentially the same. Contingency (future) ampere values were adjusted, providing increased margin.

These changes did not affect the battery sizing calculations because the methodology uses the maximum value occurring any minute for the entire minute. The total ampere-minutes within adjacent time segments were balanced and remain bounded in the existing surveillance procedures. These changes were minor (almost editorial) and do not impact the ability of the safety-related batteries to provide reliable DC power to their respective loads for the time required.

- LDCN-FSAR-97-039 (SE 97-056)

This LDCN provided for a revision to the FSAR to delete reference to the chlorine exhaust system. This system has never been used and chlorine has never been stored in the chlorine room. The exhaust ducting will be used for new weld booths in the machine shop.

Safety Evaluation Summary

It was concluded from the safety evaluation that the entire exhaust system is Quality Class G and is located in the non-power block part of the service building (machine shop). The deletion of the reference to the chlorine room is required for the FSAR to accurately reflect the current plant configuration.

This change has no impact on any safety related or important to safety structure, system or component.

- **LDCN-FSAR-97-042** (SE 97-070)

This LDCN provided for a revision to the FSAR to reflect removal of reactor core isolation cooling system valves RCIC-V-64, RCIC-V-76, RCIC-V-110 and RCIC-V-113 from a list of valves having transfer and control switches on the remote shutdown panel. These valves were removed from the design of the remote shutdown panel in 1983, however, the FSAR was not updated to reflect the change.

Safety Evaluation Summary

It was concluded from the safety evaluation that this activity only involves a revision to the FSAR to reflect the current plant configuration. This change has no impact on the design function of the reactor core isolation cooling system. The valves were removed from the remote shutdown panel design with General Electric concurrence because they were not required to comply with General Design Criterion 19.

The valves were originally included in the General Electric design to enable operators to open the valves for reactor core isolation cooling system operation if they were closed when a control room evacuation was necessary. The valves are normally open during operation and the only event that could cause them to close, coincident with the need to evacuate the control room, is a control room fire which could affect the control circuitry. However, since the reactor core isolation cooling system is not part of the dedicated fire protection shutdown system, it was not necessary to provide transfer switch controls on the remote shutdown panel in the event of a fire in the control room.

- **LDCN-FSAR-97-044** (SE 97-061)

This LDCN provided for a revision to the FSAR to reflect a change to the description of bounding fuel pool cooling events and allow for fuel movement if the resulting peak spent fuel pool temperatures do not exceed acceptance criteria.. This change will allow shorter reactor core cool down times and earlier fuel movement, provided engineering analyses are performed to validate that design limits for spent fuel temperature are not exceeded.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is no impact on safety-related systems or components by this change. Engineering analyses will validate that design limits for spent fuel temperature are not exceeded. This proposed activity does not revise any of the peak pool acceptance criteria. The activity also does not impact the design intent pertaining to operation of the spent fuel pool.

- LDCN-FSAR-97-046 (SE 97-064)

This LDCN provided for a revision to the FSAR to reflect removal of specific up-travel stop requirements on the fuel preparation machine. In addition, this LDCN removed the limitation on distance for water shielding requirements to allow setting of the up-level stop to be based on the evolution being performed and for ALARA considerations. It was stated in the FSAR that the fuel preparation machine has a permanently installed up-travel stop to prevent rising fuel above the safe water level shield level. The up-travel stops are fastened in place with screws and are adjustable.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is not a problem with allowing for high mechanical up-travel stop settings. The allowance to set the up-travel stop so that the fuel preparation machine would stop with a fuel bundle closer to the water surface is consistent with other FSAR sections pertaining to fuel handling evolutions.

This proposed activity does not increase the probability of occurrence or increase the consequences of a previously evaluated accident or transient. This activity is also bounded by the previously analyzed fuel drop accident.

- LDCN-FSAR-97-063 (SE 97-072)

This LDCN provided for a revision to the FSAR to reflect a change in the circulating water system corrosion control agent from CL-1254 to PCL-8125. Corrosion inhibitor CL-1254 is an incorrect reference to a product never manufactured by Calgon and has never been used at WNP-2. Corrosion inhibitor PCL-8125 has been used at WNP-2 since 1985.

Safety Evaluation Summary

It was concluded from the safety evaluation that this correction does not change the associated analysis of circulating water system in-leakage on the performance of the condensate demineralizers. The change is bounded by current analysis.

The data used to analyze condensate filter demineralizer performance during a condenser tube leak also remains unchanged.

- LDCN-FSAR-97-143 (SE 97-092)

This LDCN provided for a revision to the FSAR to reflect new chemicals for circulating water system corrosion control agents and eliminate reference to Calgon products. The replacement chemicals are currently used as part of the brand name Calgon product. This change will allow direct addition of the active ingredients to the circulating water system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed changes to chemical treatment for the circulating water system do not alter previously performed analyses. Changing the corrosion control agents does not increase the risk of accidents, malfunctions of equipment or alter the basis for any technical specification.

The proposed change does not alter the associated analysis of circulating water system in-leakage on the performance of the condensate demineralizers. The change is bounded by current analysis. The data used to analyze condensate filter demineralizer performance during a condenser tube leak also remains unchanged.

- LDCN-FSAR-97-151 (SE 97-096)

This LDCN provided for a revision to the FSAR to reflect analyses performed in support of continued operation with a wetwell air space temperature of up to 150 degrees Fahrenheit. These analyses included containment response to a station blackout and anticipated transient without scram events, containment peak pressure conditions, and containment negative pressure conditions. In addition, containment structural integrity and equipment qualification of impacted structures, systems and components were also evaluated. Temperature increases are due to seat leakage past the main steam relief valves. The wetwell pool temperature limit remains at 90 degrees Fahrenheit.

The FSAR was also changed to reflect current analyses which demonstrate containment atmosphere control system operability with wetwell air space temperature in excess of 95 degrees Fahrenheit. In addition, the calibration frequency for containment atmosphere control system instrumentation was extended to 24 months.

Safety Evaluation Summary

It was concluded from the safety evaluation that increasing the limiting wetwell air space temperature to 150 degrees Fahrenheit does not affect the safety function of the structures, systems and components involved. It was previously noted in the FSAR that wetwell air space temperature is normally less than 95 degrees Fahrenheit. However, when seat leakage past main steam relief valves occurs during normal power operation, increased main steam relief valve tailpipe temperature can cause the wetwell air space temperature to exceed 95 degrees Fahrenheit.

It was concluded from the evaluation that 150 degrees Fahrenheit is the maximum acceptable temperature in the wetwell air space with regard to equipment qualification, structural considerations and containment response analysis.

There is also no impact on the operability of the containment atmosphere control system. The system remains capable of performing its required safety function, provided wetwell air space temperature does not exceed a limiting value that is determined based on containment atmosphere control blower flow capacity.

Revising the calibration period to 24 months for containment atmosphere control system instrumentation is consistent with the Technical Specifications. No setpoint changes are required due to conservatisms in the associated setpoint calculations and margins associated with existing setpoints.

- **LDCN-FSAR-97-176** (SE 97-0134)

This LDCN provided for a revision to the FSAR to reflect several changes to the fire protection engineering evaluations associated with the penetration seal upgrade project. The penetration seal upgrade project has been correcting penetration seal and fire barrier problems. Changes included the addition of Thermo-Lag to the list of combustibles, revision of fire area descriptions, and revision to hourly fire ratings.

Safety Evaluation Summary

It was concluded from the safety evaluation that fire area boundaries must be capable of maintaining a sufficient hourly fire rating to prevent the worst case fire from spreading to an adjacent fire area. The proposed changes do not lower the ability of safety systems to perform during accident conditions; have no impact on the design, material or construction standards of equipment which could initiate accidents; will not cause fire barriers or adjacent equipment to be operated outside of design or testing limits; or will not change system interfaces.

The proposed fire area boundary changes still ensure fires will be limited to a single fire area in accordance with regulatory criteria and commitments. The proposed change is in conformance with Operating License Condition 2.C (14), "Fire Protection Program (Generic Letter 86-10)."

- LDCN-FSAR-97-178 (SE 97-137)

This LDCN provided for a revision to the FSAR to reflect restoration of the reactor core isolation cooling system operational and design requirements that were contained in the original FSAR and subsequent updates. In 1985, the system was reclassified as non-safety related. This change reflects system restoration to safety-related status.

Safety Evaluation Summary

It was confirmed from the safety evaluation that the changes made to the FSAR, to restore the reactor core isolation cooling system operational and design requirements reflective of its safety-related status, were consistent with original FSAR and design requirements.

The proposed change does not represent an unreviewed safety question or a change to the Technical Specifications.

- SCN 96-036 (SE 96-064)

This SCN provided for a revision to the FSAR to enhance the description of containment purge and supply isolation valves, and correct a mis-statement that the valves are designed to be locked closed in the main control room.

Safety Evaluation Summary

It was concluded from the safety evaluation that the containment purge and supply purge system is not an accident initiator and the FSAR text changes would not impact any accident analysis. The first text change was editorial in scope and the words, "that are fully qualified to close

under accident conditions," were added to the description of the containment purge and supply isolation valves. This change provides a more detailed description of the design function of these valves.

The second text change was to remove the words, "designed to be locked in the main control room and are," from a sentence describing the isolation valves. These valves were not designed to be locked closed, nor have they ever been locked closed. There is no design requirement for the valves to be locked closed. The valves will perform their design safety function of closing to isolate on a LOCA signal and are not required to be locked closed to perform this function. The current configuration of these containment isolation valves satisfies design safety function requirements.

2.6.4 Licensee Controlled Specification Changes

This section contains information pertaining to Licensee Controlled Specification (LCS) changes and is included pursuant to 10 CFR 50.59. It should be noted that Licensing Document Change Notice LDCN-LCS-97-017 represented a complete revision to the LCS manual. This revision was made to support implementation of the Improved Technical Specifications which were issued during March 1997.

- LDCN-LCS-97-017 (SE 97-018)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.0, "Requirements for Operability (RFO) Applicability," to allow for declaring supported equipment inoperable instead of only requiring that the plant operating mode be changed to another mode or specified condition. In addition, the requirement to initiate a plant shutdown on a time-specific basis was replaced with the requirement to initiate a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change would not result in any change to the plant configuration. The change impacts responses to instances where 1) a requirement for operability and associated compensatory measure are not met, 2) no compensatory measure is specified, or 3) associated compensatory measures specify entry into RFO 1.0.3. Instead of only placing the plant in another mode or specified condition in such situations, equipment could simply be considered inoperable upon entry into RFO 1.0.3.

Entry into RFO 1.0.3 is not credited in any previously evaluated transient or accident. The additional requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- LDCN-LCS-97-017 (SE 97-019)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.2.1, "Control Rod Block Instrumentation," to allow deferral of channel functional tests and channel calibrations until 12 hours after entering Operational Mode 1 from Operational Mode 2. The Operational Mode 5 surveillance testing of the average power range monitor inoperative and upscale trips, and scram discharge volume high level and scram bypass functions were eliminated. The action time for placing inoperable control rod block instrumentation channels in the trip condition was changed from one hour to seven days. The calibration frequency for the source range monitor, average power range monitor, intermediate range monitor and reactor recirculation system flow monitoring instrumentation was changed to be consistent with reactor protection system instrumentation calibration requirements.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is no design basis accident that takes credit for rod block signals. Neither the rod block nor the failure of a rod block is an initiator of an accident. The configuration and method of operation of control rod block instrumentation are not impacted by these changes.

The changes are limited to addition of standardized surveillance tests, elimination of unnecessary surveillance tests, alteration of surveillance frequencies, improvements in timing to allow elimination of the use of jumpers during surveillance testing, and adjustment of a required completion time to eliminate unnecessary placement of inoperable instrument channels in the tripped condition.

Surveillance testing will continue to be performed using methodologies and frequencies shown by experience to be effective in identifying inoperable equipment.

- LDCN-LCS-97-017 (SE 97-031)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.3.1, "Post Accident Monitoring," to reflect extension of the completion time for restoring the number of operable channels required from 48 hours to either seven days or 30 days, and from 72 hours to seven days. Compensatory measures were also changed from a plant shutdown or issuance of a special report to initiation of a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that the post accident monitoring instrumentation is not involved as an initiator in any previously analyzed accident or transient. The post accident monitoring components impacted by these changes are not credited as Regulatory Guide 1.97, Category A or Type I instruments. The loss of these instruments would not impact offsite doses.

The proposed change will provide additional time to restore inoperable post accident monitoring instrumentation to service. The requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- LDCN-LCS-97-017 (SE 97-013)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.3.1, "Diesel Generator Volt Meters," to reflect a change in the calibration frequency from every refueling outage (or 12 months) to every 24 months for the emergency diesel generator volt meters and ammeters.

Safety Evaluation Summary

It was concluded from the safety evaluation that extending the surveillance frequency to 24 months would still be within the vendor-recommended frequency. Reviews of historical maintenance and surveillance data have shown that these tests normally pass the surveillances at the previous frequencies. An evaluation was performed using this data and it was concluded that extending the frequency would have no impact on the safety function of systems or components.

This is also consistent with Generic Letter 94-01, "Removal of Accelerated Testing and Special Reporting Requirements for Emergency Diesel Generators," and to accommodate a change to the WNP-2 maintenance cycle from 12 months to 24 months. In Generic Letter 94-01, the NRC advised all licensees that extending an 18-month surveillance interval to 24 months could be justified on the basis of a successful equipment performance history.

- LDCN-LCS-97-017 (SE 97-020)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.4.7, "Reactor Coolant System Interface Valves Leakage Pressure Monitors," to reflect a change in a compensatory measure in the event an inoperable reactor coolant system interface valve monitor cannot be restored to operable status within 30 days. The compensatory measure was changed from a plant shutdown to initiation of a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that this proposed change has no impact on any previously analyzed accident or transient. No automatic functions are initiated by the instruments. The alarm function simply notifies the operator of a potential problem. The proposed change will not result in any physical change to the plant.

The proposed change is to require initiation of a problem evaluation request within 24 hours of discovery of unacceptable reactor coolant system interface valve leakage, in lieu of a shutdown after 30 days of discovery of the leakage. The requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- **LDCN-LCS-97-017** (SE 97-021)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.5.2, "Automatic Depressurization System ADS Inhibitor," to clarify the impact on automatic depressurization system operability in the event of an inoperable inhibit switch. In addition, the inhibit switch surveillance frequency was changed from three to 24 months and the method of testing was changed from a channel functional test to a logic system functional test.

Safety Evaluation Summary

It was concluded from the safety evaluation that loss of the automatic depressurization system inhibit switch function was a non-significant risk contributor to core damage frequency and offsite release. The inhibit switch is an operational function and is not credited for any accident or transient. Therefore, there is no increase in the probability of an accident or transient previously evaluated in the Safety Analysis Report.

The change will allow continuing plant operation in the event of an inoperable manual inhibit switch, provided that the inoperable switch has not resulted in inhibiting the automatic operation of the automatic depressurization system. The design configuration and method of operation will not be changed by this proposed activity. The proposed change in surveillance testing imposes more complete testing on the circuitry such that all contacts and functions are fully tested.

The surveillance extension is consistent with the guidance in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." Reviews of historical maintenance and surveillance data have shown that these tests normally pass the surveillances at the previous frequency. An evaluation was performed using this data and it was concluded that extending the frequency would have no impact on the safety function of systems or components.

- **LDCN-LCS-97-017** (SE 97-032)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.5.3, "RCIC Isolation Instrumentation," to reflect that the trip setpoint was relocated to plant procedures, the allowable value for the trip channel calibration was changed from 1.85 psig to 1.88 psig, the logic system functional test frequency was changed from 18 months to 24 months, and the response time surveillance test of isolation valve closure cycle was deleted.

Safety Evaluation Summary

It was concluded from the safety evaluation that these changes involve details of surveillance testing and have no impact on plant configuration or method of operating plant systems. Surveillance testing will continue to be performed in accordance with approved plant processes and procedures to verify operability and functionality of reactor core isolation cooling system isolation instrumentation.

The proposed changes remain within established design and safety limits. The surveillance extension is consistent with the guidance in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." Reviews of historical maintenance and surveillance data have shown that these tests normally pass the surveillances at the previous frequency. An evaluation was performed using this data and it was concluded that extending the frequency would have no impact on the safety function of systems or components.

- LDCN-LCS-97-017 (SE 97-022)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.7.3, "Explosive Gas Monitoring System," to reflect a change in a compensatory measure in the event an inoperable hydrogen monitor is not restored within the required time frame. The compensatory measure was changed from issuance of a special report to initiation of a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change will not result in any physical change to the plant or method of operating plant systems. The change only impacts the administrative response to a failure to restore an inoperable hydrogen monitor to operable status.

The requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- LDCN-LCS-97-017 (SE 97-012)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.3.7.6, "Turbine Overspeed Protection," to reflect changes in the turbine valve testing frequency from monthly to quarterly.

Safety Evaluation Summary

It was concluded from the safety evaluation that the main turbine manufacturer (Westinghouse) performed a probabilistic safety analysis pertaining to turbine valve test frequency. As a result of the analysis, it was concluded by the manufacturer that the test frequency for governor, throttle, intercept and reheat valves can be performed on a quarterly basis.

Nuclear Regulatory Commission acceptance criteria states that the annual probability of turbine missile ejection should not exceed 1.0E-5 per year for turbines such as the one at WNP-2. The Westinghouse analysis determined that the annual probability of turbine missile ejection from a BB-296 turbine to be 8.8E-7 per year.

Destructive overspeed is assumed to occur when a system separation occurs and at least one governor valve and one throttle valve in the same steam chest fail to close. Based on previous Westinghouse reports, the design and intermediate overspeed failure probabilities are not major contributors to turbine missile probability for BB-296 turbines. This change only extends the current surveillance frequency to cycle each valve from monthly to quarterly. There are no changes to plant configuration or the originally-postulated consequences of destructive turbine overspeed.

- LDCN-LCS-97-017 (SE 97-023-00 and 97-023-01)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.4.1, "RCS Chemistry," to reflect an increase in the sampling intervals pertaining to monitoring of the reactor coolant.

Safety Evaluation Summary

Based on operational history, it was concluded from the safety evaluation the increase in sampling intervals and completion times were adequate to prevent long-term degradation of the reactor coolant pressure boundary. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is unchanged.

Monitoring is not used to ensure that a process variable used as an initial condition for a design basis accident is maintained within limits. Reactor coolant chemistry is also a non-significant risk contributor to core damage frequency and offsite release. Increasing the monitoring intervals and completion times will not result in an increase in the radiological dose at the site boundary.

- LDCN-LCS-97-017 (SE 96-107)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.4.1, "RCS Chemistry," to reflect a change in the frequency of reactor coolant grab sampling for pH and chloride from 72 hours to seven days. In addition, the frequency of grab samples was changed from every 8 hours to every 24 hours when the conductivity monitor is inoperable. [This was subsequently superseded by Safety Evaluations 97-023-00 and 97-023-01 (LDCN-LCS-017).]

Safety Evaluation Summary

It was concluded from the safety evaluation that the limits for reactor water chemistry were not changed and the reactor is maintained within the current boundaries for operation. The sampling frequency is changed to take credit for conductivity monitor operation and increased sampling frequency when the monitor is not in service.

Based on operational history, it was concluded from the safety evaluation the increase in sampling intervals and completion times were adequate to prevent long-term degradation of the reactor coolant pressure boundary. Therefore, the probability of occurrence of a malfunction of equipment important to safety previously evaluated is unchanged.

Monitoring is not used to ensure that a process variable used as an initial condition for a design basis accident is maintained within limits. Reactor coolant chemistry is also a non-significant risk contributor to core damage frequency and offsite release.

- LDCN-LCS-97-017 (SE 97-024)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.6.1.5, "Suppression Pool Spray," to reflect a change in compensatory measures in the event of an inoperable train of suppression pool spray. The compensatory measure was changed from being in either hot or cold shutdown, to initiation of a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change will not result in any physical change to the plant or method of operating plant systems. Suppression pool spray is not credited in any design basis accident. The suppression pool spray system is a defense-in-depth accident mitigating feature that has no role in accident prevention or the protection of important to safety equipment.

The requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- **LDCN-LCS-97-017** (SE 97-025)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.7.1, "Area Temperature Monitoring," to reflect a change in a compensatory measure in the event where area temperatures exceed prescribed limits. The compensatory measure was changed from issuance of a special report to initiation of a problem evaluation request.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change will not result in any physical change to the plant or method of operating plant systems. The change only impacts the administrative response to a failure to restore area temperatures to within prescribed limits.

The requirement to initiate a problem evaluation request assures timely resolution of the problem and that appropriate management attention is directed toward the issue.

- **LDCN-LCS-97-017** (SE 96-073)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.8.4, "24 VDC Sources," to reflect several changes pertaining to operating shutdown requirements, compensatory measures, minimum battery terminal voltage, minimum battery connection resistance, battery charger current capacity test method, battery charger current capacity test interval and battery service test interval.

Safety Evaluation Summary

It was concluded from the safety evaluation that the 24-volt batteries are a support system for the source range monitors and intermediate range monitors. These instruments are not assumed to function during or after a design basis loss of offsite power.

The batteries, chargers and supported instruments are not involved as an initiator in any accident or transient previously evaluated. Upon a loss of offsite power, the reactor will scram and all control rods will insert.

- **LDCN-LCS-97-017** (SE 97-010)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.8.4, "24 VDC Sources," to reflect a change in the battery charger and battery capacity surveillance from 18 months to 24 months. In addition, the test duration was changed from a constant load test at 100 percent for four hours, to three separate and sequential tests at 50 percent, 75 percent and 100 percent loading for 30 minutes at each load rating, for a total of 1.5 hours.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed changes remain within established design and safety limits. The surveillance extension is consistent with the guidance in Generic Letter 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle." Reviews of historical maintenance and surveillance data have shown that these tests normally pass the surveillances at the previous frequency. An evaluation was performed using this data and it was concluded that extending the frequency would have no impact on the safety function of systems or components.

The battery charger load test was changed to reflect vendor recommendations. The battery vendors recommend a charger load test that step-loads the battery chargers at three distinct loads, not just at the 100 percent rating.

- LDCN-LCS-97-017 (SE 97-027)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.8.6, "24 VDC Battery Parameters," to reflect deletion of the requirement to verify the average electrolyte temperature within seven days after a battery discharge or overcharge. This change allows for a temporary increase in the electrolyte level during and following an equalize charge, and allows a charging current of less than two amps when on a float charge for meeting specific gravity limits following a battery charge.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed changes represent accepted industry practice and do not result in the loss of margin. The 24-volt batteries are a support system for the source range monitors and intermediate range monitors. These instruments are not assumed to function during or after a design basis loss of offsite power.

The batteries are not involved as an initiator in any accident or transient previously evaluated. The proposed changes will not result in a decrease in the reliability of the batteries.

- LDCN-LCS-97-017 (SE 97-028)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.8.9, "AC Circuits Inside Primary Containment," to reflect an increase in the completion time from one hour to four hours to deenergize the required electrical circuits to protect primary containment penetrations.

Safety Evaluation Summary

It was concluded from the safety evaluation that adequate protection of the primary containment penetrations is maintained. Maintenance and operational history has demonstrated that the chance is small that an electrical overload would fail a primary containment penetration in the additional three hours. It has also been determined that the loss of this protection is a non-significant risk contributor to core damage frequency and offsite release.

Protection of the containment penetrations is an accident mitigation function. Extending the allowable time to achieve electrical isolation of a penetration would have no impact on accidents or transients previously evaluated.

- **LDCN-LCS-97-017** (SE 97-029)

This portion of LDCN-LCS-97-017 provided for a revision to the LCS, Section 1.8.10, "Primary Containment Overcurrent Protection," to reflect elimination of the requirement to initiate a plant shutdown as a compensatory measure in the event one or more primary containment penetration overcurrent devices become inoperable.

Safety Evaluation Summary

It was concluded from the safety evaluation that the primary containment penetration overcurrent protection function or devices are not identified as an initiator of any previously identified accidents or transients. The loss of these protective devices has been determined to be a non-significant risk contributor to core damage frequency and offsite release.

This change only impacts the administrative response to an inoperable overcurrent device. The administrative response to an inoperable overcurrent device is to deenergize the affected circuit. This can always be accomplished from outside primary containment during any operating condition. Therefore, the fall-back compensatory measure of initiating a plant shutdown is not needed and represents an unnecessary operational challenge.

Furthermore, if a penetration were damaged, the limiting condition for operation of Technical Specification 3.6.1.1, "Primary Containment," would govern.

- **LDCN-LCS-97-076** (SE 97-076)

This LDCN provided for a change to the LCS Section 1.3.3.2, "Remote Shutdown System," to reflect that two tables of remote shutdown instrumentation and remote shutdown functions were combined into one table since both are comprised of the same instrumentation used to support operator action to achieve safe shutdown. A diesel generator voltmeter and frequency meter were moved to a new equipment status monitoring table. New requirements for operability and

surveillance requirements were added to ensure that operability of the instruments associated with remote shutdown equipment status monitoring is maintained. Administrative clarifications were also made to this section.

Safety Evaluation Summary

It was concluded from the safety evaluation that the remote shutdown system is not assumed to be operable for any design basis accident previously evaluated, but the system is required to be operable to ensure the plant complies with General Design Criterion 19. These proposed changes do not result in any physical change to the plant or method of operating plant systems.

The change also adds additional monitoring components for remote shutdown and places more restrictive testing requirements on the instruments added.

- LDCN-LCS-97-093 (SE 97-081)

This LDCN provided for a change to the LCS Section 1.3.7.6, "Turbine Overspeed Protection System," to allow for startup of the main turbine to perform operability surveillance testing on the main turbine valves. This change will allow for testing of those valves that cannot be tested until reactor steam flow is sufficient to open the valves.

During startup of the main turbine, all of the valves are routinely tested in accordance with plant procedure 2.5.7, "Main Turbine," to ensure they will be in a closed position on a turbine trip. Also, during turbine roll-up to 1800 rpm, the throttle and governor valves are required to function properly to control turbine speed with the intercept and reheat valve open. After the generator is initially synchronized, all of the turbine valves are open, except the one and four governor valves. These valve will start to open at approximately 65 percent power when there is sufficient steam flow.

Safety Evaluation Summary

It was concluded from the safety evaluation that this proposed change does not increase the probability of occurrence of a turbine overspeed event. The proposed change does not result in any physical change to the plant. Testing of the turbine valves will ensure that the valves will close when required during a turbine trip and, therefore, avoid an overspeed condition. This proposed change does not involve an increase in the probability or consequences of an accident previously evaluated because the probability of occurrence does not change, does not create the possibility for an accident or malfunction of a different type because the existing design is not changed, and does not involve a significant reduction in the margin of safety for any technical specification since the turbine overspeed system is not addressed in the Technical Specifications.

2.6.5 Plant Tests and Experiments

This section contains information pertaining to tests and experiments and is included pursuant to 10 CFR 50.59.

PMR 87-0244-OG (SE 93-200 and SE 97-078)
Procedure 8.3.339

On March 27, 1997 the Supply System performed a reactor feedwater pump trip test to demonstrate that the Digital Feedwater Level Control (DFWLC) System and Reactor Recirculation Adjustable Speed Drive (ASD) System modifications would function to avoid a low water level (Level-3) scram. This test was essentially a repeat of the original startup test for a single feedwater pump trip and was being conducted to confirm the integrated performance capability of the DFWLC and ASD modifications.

During the initial phase of the test, the plant responded as expected with a reactor recirculation system pump runback to 27 Hz upon occurrence of a low water level alarm (Level-4), coincident with a feedwater pump trip and with the DFWLC system responding to restore water level. However, there was an unexpected and additional runback of the reactor recirculation system pumps to 15 Hz.

The unplanned runback was initiated by a steam dome to recirculation system differential temperature cavitation interlock signal that existed for greater than 15 seconds. This interlock is a feature of the BWR-5 design and was not anticipated to actuate during test conditions. The actuation had not occurred during original plant startup testing and the potential for its occurring was not considered in the pre-test evaluations.

The second runback appeared to have placed the reactor into Region A of the power-to-flow map. Accordingly, consistent with pre-test planning, plant procedures and the Technical Specifications, the operators manually scrammed the reactor. (It was later determined by follow-up engineering analyses that Region A had not been entered.)

As expected in response to the manual scram, reactor vessel water level decreased to just below +13 inches. In response to the low level condition, reactor feedwater flow increased and level was recovered to above +13 inches. During this evolution and at approximately +18 inches, Reactor Feedwater Pump RFW-P-1A speed failed to decrease in response to the increasing level control signal. As a result, reactor vessel water level reached +54 inches (Reactor Vessel Water Level - High: Level 8) and the pump tripped on high level.

Plant operators successfully returned both reactor feedwater pumps to operation and restored level to within normal limits.

Safety Evaluation Summary

A new safety evaluation (SE 97-078) was performed to provide the bases for determining that the plant response to the March 27, 1997 test did not reveal any information that would invalidate the original design change safety evaluation (SE 93-200), or result in an unreviewed safety question. For the purpose of the unreviewed safety question determination, the activity was defined as the plant response to the reactor feedwater pump trip test event. This included the differential temperature cavitation interlock trip and associated reactor recirculation pump runback, apparent indication of entry into Region A of the power-to-flow map, and water level response. The new safety evaluation is summarized as follows.

Initiation of the differential temperature cavitation interlock and subsequent runback of the reactor recirculation system pumps to 15 Hz is bounded by the more severe and previously-analyzed trip of two recirculation pumps transient.

Reactor recirculation system flow run-back and recirculation pump trip events that lead to entry into Region A of the WNP-2 power-to-flow map were considered in establishing the stability region boundaries. Personnel recognized during the development of the stability region that certain unplanned operational occurrences, most notably recirculation pump trips and runbacks, would lead to entry into the stability region. The region definitions fully account for entry into the region as a result of a core flow reduction, independent of the probability of occurrence of such a reduction in core flow.

Entry into Region A of the power-to-flow map is controlled by Technical Specification 3.4.1, "Recirculation Loops Operating." Compliance with the limiting condition for operation action statements in this specification, in the event of entry into Region A, assures that an unreviewed safety question does not exist. Although the Technical Specifications allow 15 minutes before action is necessary, management expectations, Operating Procedure PPM 4.12.4.7, "Unintentional Entry into Region of Potential Core Power Instabilities," and conservative control room decision making resulted in a manual scram upon the apparent entry into Region A. (The reactor was scrammed within approximately two minutes following the unplanned runback to 15 Hz.) This precluded operation in the exclusion region.

The adjustable speed drive and digital feedwater level control system pre-scram response to the feedwater pump trip and subsequent recirculation flow runback was as expected, with regard to reactor vessel water level, and did not result in a Level-3 scram. As feedwater flow stabilized and prior to the manual scram, reactor vessel level swelled and peaked at slightly over 51 inches, avoiding a Level-8 isolation. The trip of a single feedwater pump, that does not result in a reactor trip, indicates that the control system response does not increase the probability of a more severe transient resulting from an operational event. Other, less limiting, operational events are analyzed

in the General Electric Adjustable Speed Drive Control System Report and are shown not to degrade due to adjustable speed drive and digital feedwater level control system response. Accordingly, less limiting transients do not become additional transients for FSAR analyses purposes.

The post-scram response was not dissimilar to what would have been seen with the previous analog system. Since the Level-8 trip was reached post-scram, there was no adverse impact on fuel thermal limits. For long-term cooling and inventory makeup, the high pressure core spray and reactor core isolation systems would be available if water level decreased to their initiation setpoints. Accordingly, the transients in the FSAR are still bounding and the consequences of an accident, as analyzed in the FSAR, were not increased.

To demonstrate that the feedwater level control system and recirculation flow runback feature would prevent a scram following the trip of a single feedwater pump during power operation, the test was successfully repeated on July 23, 1997 when the plant returned to stable, full power operation following the R-12 Maintenance and Refueling Outage. The systems responded as designed.

2.6.6 Plant Procedure Changes

The section contains information pertaining to Plant Procedure Manual (PPM) changes and is included pursuant to 10 CFR 50.59.

- PPM 2.4.3 (SE 97-039)

  This procedure provides detailed instructions for operation of the low pressure core spray system. The procedure was revised to provide for an alternate means to fill and repressurize the low pressure core spray system in the event that normal keep fill pressure from the keep fill system is unavailable. The alternate makeup water source can be from either the demineralized water system or the residual heat removal system, Loop A, by means of a temporary hose connection.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that the time the low pressure core spray system would be out of service for this evolution is bounded by the complete failure of the emergency core cooling system (Division 1) transient that is postulated in the FSAR. Restoration of the system to service will ensure that its design function as an accident mitigation system is available. The proposed change also does not create the possibility or increase the consequences of a malfunction of equipment important to safety described in the FSAR.

  Existing flooding and moderate energy line break analyses are bounding in the unlikely event of a hose rupture should it occur during the filling evolution. Restoration of the low pressure core spray system to operable status will allow the system to perform its design function. The allowable outage time of seven days for the system more than bounds the time required to fill, vent and pressurize the system.

- PPM 2.4.4 (SE 97-040)

  This procedure provides detailed instructions for operation of the high pressure core spray system. The procedure was revised to provide for an alternate means to fill and repressurize the high pressure core spray system in the event that normal keep fill pressure from the keep fill system is unavailable. The alternate makeup water source can be from either the demineralized water system or the residual heat removal system, Loop C, by means of a temporary hose connection.

Safety Evaluation Summary

It was concluded from the safety evaluation that the time the high pressure core spray system would be out of service for this evolution is bounded by the complete failure of the emergency core cooling system (Division 3) transient that is postulated in the FSAR. Restoration of the system to service will ensure that its design function as an accident mitigation system is available. The proposed change also does not create the possibility or increase the consequences of a malfunction of equipment important to safety described in the FSAR.

Existing flooding and moderate energy line break analyses are bounding in the unlikely event of a hose rupture should it occur during the filling evolution. Restoration of the high pressure core spray system to operable status will allow the system to perform its design function. The allowable outage time of 14 days for the system more than bounds the time required to fill, vent and pressurize the system.

- PPM 2.8.6A (SE 97-048)

This is a new procedure which was created to provide instructions for reducing the water level in the condensate storage tanks below the normally maintained level for maintenance, expanded range of inventory use during refueling outages, or other infrequent needs.

Safety Evaluation Summary

It was concluded from the safety evaluation that this activity will not increase the probability or consequences of an accident or transient previously evaluated in the FSAR. The condensate storage and transfer system is a non-safety related water source that is not credited during accident conditions. Furthermore, the procedure will only be authorized for use when the plant is in a cold shutdown, refueling outage or defueled condition.

Cold shutdown, refueling outage and defueled operational inventory and interlock requirements are maintained by this change.

- PPM OSP-INST-H101 (SE 97-099)

This procedure provides instructions for demonstrating the operability of each instrument channel by comparison of all channel indications or comparison of other independent instrumentation measuring the same variables as required by the Technical Specifications. The procedure was revised to allow an alternate method of monitoring drywell identified leakage rate when the normal flow rate instrument (EDR-FT-37) is out of service. The alternate method uses other existing control room instrumentation to verify that total leakage surveillance requirements are satisfied.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed alternate method for determining drywell identified leakage provides an acceptable method of satisfying the Technical Specifications and verifying that the limits on total leakage are maintained. The equipment used by the alternate method are of the same qualification level as the plant equipment credited during normal operation.

The proposed method for determining drywell identified leakage does not reduce the margin of safety as defined in the Technical Specifications for operational leakage, nor does it conflict with Regulatory Guide 1.45 guidance pertaining to reactor coolant pressure boundary leakage detection systems.

- PPM OSP-RPV-R801 (SE 97-003)

This procedure provides instructions for performing a pressure test on ASME code piping and components to satisfy inservice and leakage pressure testing requirements. This is a new replacement procedure that was created to reflect implementation of the Improved Technical Specifications. The change to the procedure that initiated the safety evaluation was a change from 212 degrees Fahrenheit to 275 degrees Fahrenheit during the reactor pressure vessel leakage test. The upper temperature restriction was removed with issuance of the Improved Technical Specifications, which also allowed for the test to be performed during Operational Mode 4, provided certain Operational Mode 3 limiting conditions for operation are satisfied.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change would not impact the mechanisms that are considered initiators for transients or accidents. The change is bounded by FSAR Chapters 6 and 15 analyses, particularly by the steam line break outside of containment accident.

The components exposed to the changed condition during the reactor pressure vessel leakage test are the isolation valves and reactor coolant boundaries. These components are passive and the increased temperature improves the strength of the material since the temperature remains to the right of the revised pressure/temperature limit curve and the components have been qualified for higher temperatures. The change in the pressure/temperature limit curve introduces a conservatism over at least the next 20 years by introducing more margin in terms of vessel metal response.

2.6.7 Miscellaneous

This section contains information pertaining to other plant activities and is included pursuant to 10 CFR 50.59.

- **Configuration Document Change Request CDCR 97-01-019 (SE 97-007)**

  This configuration document change request provided for the removal of the setpoint values for all non-safety circuits on the FSAR logic control diagrams for the service water system. There is no requirement to provide control logic diagrams for non-safety circuits.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that there is no impact on the service water system or the ability of the service water system to perform its design function. The proposed activity is a drawing change only. The safety-related circuits on these drawings remain unchanged by this activity.

- **Configuration Document Change Request CDCR 97-06-017 (SE 97-098)**

  This configuration document change request provided for a correction to an extraction steam system drawing to reflect current plant configuration. Existing extraction steam system lines, seal steam system line tie-ins and extraction steam system connections to the condensate feedwater heaters were incorporated into the drawing.

  Safety Evaluation Summary

  It was concluded from the safety evaluation that there is no impact on the extraction steam or seal steam systems or the ability of the extraction steam and seal steam systems to perform their intended design functions. The proposed activity is a drawing change only. The Quality Class 2, Seismic Class 2, extraction steam and seal steam systems have no analyzed impact on any safety-related installation.

- **Configuration Document Change Request CDCR 97-07-004 (SE 97-084)**

  This configuration document change request provided for a revision to a containment instrument air system drawing to reflect a normally-closed position for valve CIA-V-738. The drawing reflected a normally-open position, while plant operating procedures require this valve to be usually closed.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is no impact on the containment instrument air system or the ability of the containment instrument air system to perform its design function. The proposed activity is a drawing change only. This valve isolates the backup feed from the control and service air system to the non-safety related portion of the containment instrument air system. Two other normally-closed valves in the line perform the containment instrument air isolation function.

- **Fire Protection Engineering Evaluation FPF 1.2** (SE 97-059)

This fire protection engineering evaluation provided for an extension for completion of the Thermo-Lag resolution project from 1997 to 1999. The Supply System formally notified the NRC of this schedule change in follow-up responses to Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers."

Safety Evaluation Summary

It was concluded from the safety evaluation that the ongoing compensatory measures of fire tours and other defense-in-depth features provide adequate assurance that post-fire safe shutdown will be achieved during the extended period of Thermo-Lag inoperability.

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

- **LDCN-LBD-97-040** (SE 97-058)

This licensing basis document change provided for the deletion of superseded Technical Specification requirements to submit special reports to the NRC in the event that the meteorological monitoring instrumentation or loose part detection system channels are inoperable. These requirements were removed from the Technical Specifications as part of implementation of the Improved Technical Specifications.

Safety Evaluation Summary

It was concluded from the safety evaluation that the requirement to file a special report for inoperable instrumentation channels is administrative and serves no function to protect the health and safety of either the public or plant personnel. There is no impact to nuclear safety by deletion of these reporting requirements.

Furthermore, the meteorological monitoring and loose part detection systems did not meet the importance criteria for inclusion into the Improved Technical Specifications, nor are these two systems included in the Maintenance Rule. A summary of meteorological monitoring system performance is also provided to the NRC by means of the annual effluent report.

- LDCN-ODCM-97-084 (SE 97-135)

This LDCN provided for a revision to the Offsite Dose Calculation Manual to reflect a reduction in sampling requirements for reactor power changes of greater than 15 percent in one hour when the fission gas release is less than 15,000 microcuries per second. In addition, sampling requirements were added for reactor power changes of greater than 15 percent in one hour when the fission gas release is greater than 15,000 microcuries per second. The change also separated the grab sampling and analysis requirements for startups and shutdowns for changes in thermal power of greater than 15 percent within one hour.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change will reduce the sampling and analysis frequency when the fission gas release rate from the core is at low levels. The reduction in sampling will reduce information available to the operator on the rate at which fission gases are released from the core. However, this reduction in information will only occur at release rates that are a small fraction of the Technical Specification limits.

For the purposes of determining offsite dose impact, all accidents and malfunctions assume that either the primary coolant is at the Technical Specification limit or that a fuel failure has occurred. This proposed change has no impact on equipment important to safety and is bounded by previously evaluated accidents and transients.

- LDCN-OQAPD-97-043 (SE 97-060)

This LDCN provide for a revision to the Operational Quality Program Description to reflect issuance of new site-wide procedures, clarification of the qualification requirements for the Quality Manager and Quality Services Supervisor, and implementation of a new process for review and approval of programs and procedures. In addition, the memberships of the Plant Operations Committee and Corporate Nuclear Safety Review Board were changed.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change to the Operational Quality Program Description does not impact important to safety components or increase the risk of an accident or malfunction. There are no new accident scenarios introduced by this change.

The change to the procedure review and approval process is administrative in scope. The change to the composition of the Plant Operations Committee and Corporate Nuclear Review Board is administrative and organizational in scope. The change to the qualification requirements for the Quality Manager and Quality Services Supervisor is also considered an administrative change. The position-specific requirements for the Quality Manager, as defined in ANSI/ANS-3.1-1978, Section 4.4.5, are maintained.

- LDCN-TSB-97-033 (SE 97-49)

This LDCN provided for a revision to the Technical Specification Bases to reflect an alternate method for performing the source range monitor signal-to-noise ratio determination. This alternate method was recommended by General Electric. The use of the alternate method eliminates the need to send personnel undervessel to manipulate source range monitor cables, which is the current method. Source range monitor signal-to-noise ratios are used to verify that source range monitors will respond should a reactivity change occur while shutdown.

Safety Evaluation Summary

It was concluded from the safety evaluation that this activity does not increase the probability or impact of equipment malfunctions. The previous process for determining the signal-to-noise-ratio compares the source range monitor with the detector fully inserted in the core to the source range monitor count rate with the detector withdrawn from the core. The count rate with the detector inserted is a combined signal and noise value. The count rate with the detector removed is due to noise. These two values are compared and a signal-to-noise ratio is determined.

The change uses a method that does not require withdrawal of the detector to determine the signal-to-noise ratio. Instead, an initial signal and noise reading is taken with the detector fully inserted in the core and normal high voltage applied to the detector. This is the normal configuration of the source range monitor while shutdown. The high voltage is then removed from the detector and an evaluation is performed to determine if there was a noise component from the high voltage power supply. If a high voltage power supply was not apparent, a second reading is taken. This reading is the noise component of the source range count rate. The initial signal and noise readings are then compared to the final noise reading and the signal-to-noise ratio is determined.

Source range monitors are neither initiators for plant accidents nor are they credited for mitigating any plant accident. The proposed change has been recommended by General Electric as a alternate method for determining the source range monitor signal-to-noise ratio.

- **Problem Evaluation Request 297-0394 (SE 97-077)**

This problem evaluation request corrective action provided for an evaluation to determine the impact of the formation of gaps between the reactor recirculation system jet pump inlet mixer and associated restrainer bracket set screws during the Cycle 13 operating cycle. Inspections had determined that jet pump set screw gaps can form during a single operating cycle.

Safety Evaluation Summary

It was concluded from the safety evaluation that operation with jet pump set screw gaps is acceptable from a structural safety perspective. Based on the results of fatigue analysis, it was concluded that jet pump structural integrity for all set screw gaps will be maintained for the current operational cycle. The accumulated fatigue usage assumed through Cycle 13 is less than the allowable fatigue usage factor. This conclusion supports operation with the potential for gap formation during Cycle 13.

- **Power-to-Flow Map (SE 96-027)**

This activity consisted of changing the definition (reducing the size) of the area of increased awareness to the region of the Cycle 12 power-to-flow map between the 54 percent rod line and 50 percent rated core flow. This applies to single and two-loop reactor recirculation system pump operation. An additional requirement was added to maintain the core average boiling boundary greater than three feet for operation between 25 percent rated power and 50 percent rated core flow.

Safety Evaluation Summary

It was concluded from the safety evaluation that the event of concern is reactor power oscillations, which is neither an accident nor a transient in the licensing basis documents. The BWR Owners Group recommendations place no restrictions on power operation below the 70 percent rod line. Therefore, the area of increased awareness is not required. However, core power oscillations can occur below the 70 percent rod line if a core power distribution is established that is sufficiently adverse to stability. Accordingly, it would be prudent to establish a region on the power-to-flow map where awareness of core stability issues is increased. The area of increased awareness continues to fulfill that function and, as such, is a conservative application of the BWR Owners Group recommendations.

Furthermore, to preclude power distributions significantly adverse to core stability, a requirement for boiling boundary is established. The BWR Owners Group recommendations do not specify this as being necessary below the 70 percent rod line. Therefore, the boiling boundary requirement is a conservative application of the BWR Owners Group recommendations.

This change continues to fulfill the intent of commitments, made in Licensee Event Report 92-037-03, "Manual Reactor Scram due to Core Instability," to maintain awareness of core stability issues and to preclude instability.

- **Relief Valve Setpoint Data Sheets SLC-RV-29A and SLC-RV 29B** (SE 97-121)

The relief valve setpoint datasheets for standby liquid control system relief valves SLC-RV-29A and SLC-V-29B were revised to allow for a relaxation of the as-found setpoint tolerances for the valves. The as-left tolerances were not changed. The new tolerances are acceptable in accordance with the inservice testing program plan.

Safety Evaluation Summary

It was concluded from the safety evaluation that the new minimum acceptable value for as-found pressure is sufficient to ensure that the standby liquid control system maintains operability and will inject sodium pentaborate solution into the reactor at design pressure.

The new tolerances are also consistent with the Improved Technical Specifications and Supply System and General Electric calculations.

- **Request for Technical Services RFTS-97-01-007** (SE 97-083)

This request for technical services provided for a clarification of the decay heat removal methods other than residual heat removal system shutdown cooling and the draining of one standby service water spray pond for maintenance.

Safety Evaluation Summary

It was concluded from the safety evaluation that alternate methods of shutdown cooling are fuel pool cooling and the reactor water cleanup system operating with the regenerative heat exchanger bypassed, or in combination with the control rod drive system or condensate system.

The draining of one pond during a refueling outage when the vessel head is off and the vessel cavity is flooded up at least 22 feet above the vessel flange will not increase the probability of occurrence of an accident because the ultimate heat sink and standby service water systems are not credited as being accident initiators.

Since the two spray ponds will be isolated from each other during this time period, the draining of one pond would not impact the decay heat removal function of the other pond and associated service water division. In addition, the consequences of an accident will not increase because no emergency core cooling systems are required to be operational during refueling when the spent fuel pool storage gates are removed and the water level is at least 22 feet above the vessel flange.

- **Technical Evaluation Request 95-0255-00 (SE 96-105)**
  Technical Evaluation Request 96-0182-00 (SE 96-105)

These technical evaluation requests provided for removal of the seal-in contacts from turbine gland sealing system (sealing steam system) valves SS-V-1A, SS-V-1B and SS-V-4 to allow for throttling capability. The purpose of the seal-in contact is to provide a complete valve stroke ("full open to close" or "full close to open") upon momentary placement of the control switch in either the close or open position. With the seal-in removed, the valve will stroke for only as long as the control switch is held in the open or close position.

Safety Evaluation Summary

It was concluded from the safety evaluation that the sealing steam system is not credited with mitigating the consequences of any previously analyzed accident. This change does not introduce an accident or malfunction more limiting than the already analyzed complete system malfunction. Furthermore, these valves are non-safety related and do not provide a safety function. The valves cannot affect, influence or degrade the function of any safety-related or augmented quality structures, systems or components.

Valves SS-V-1A or SS-V-1B could cause the loss of the sealing steam system only if the valve supplying the system at the time were to be inadvertently shut. Valve SS-V-4 could cause loss of adequate supply to the sealing steam system only if the valve were to be inadvertently shut while the auxiliary boiler is supplying sealing steam. With the seal-in contact in place, inadvertent closure of the valve could happen due to any instantaneous closure of the close contact. However, with the seal-in contact removed from the circuit, inadvertent valve closure would require the close contact to be held long enough for the valve to complete its stroke.

- **Technical Evaluation Request 96-0193 (SE 97-089)**

This technical evaluation request provided for plant process computer points for reactor water cleanup system inlet and outlet flow signals. This change adds a resistor to each of the flow instrument loops, across which the process computer will monitor voltages to derive flow values. The addition of the computer points will make it possible to remotely read these flow signals, as well as locally in the control room.

Safety Evaluation Summary

It was concluded from the safety evaluation that the two new computer points will have no impact on the function or operation of the reactor water cleanup system. There are no specific regulatory requirements associated with the plant process computer. Electrical separation criteria and routing requirements are maintained to ensure that safety-related circuits are isolated from non-safety related circuits.

The resistors are qualified for safety-related service. The small increase in circuit resistance created by the new resistors are not of concern and would be compensated by the surveillances used to perform post-installation instrument loop calibrations.

- **Technical Evaluation Request 96-0213 (SE 97-041)**

This technical evaluation request provided for the replacement of maintenance flanges with socket weld couplings on two, three-quarter-inch piping lines in the reactor recirculation system. This change was made to reduce the possibility of socket weld fatigue failures due to steady state vibration of the reactor recirculation system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the replacement is permitted by applicable WNP-2 piping design specifications. There is no impact on safety-related systems or components. The two, three-quarter-inch piping lines impacted by this activity are used for venting or draining purposes only. Installation of socket weld couplings will not change the design function or the pressure and flow ratings for these lines.

- **Technical Evaluation Request 97-0044 (SE 97-057)**

This technical evaluation request provided for the replacement of temperature control units TSW-TC-9 and TSW-TT-9 in the plant service water system with a new component (TSW-TIC-9). The existing components provide input to temperature control valve TSW-TCV-9,

which controls cooling of hydrogen in the main generator. The new equivalent component replaces the two existing components. The unit will measure the hydrogen temperature in the main generator and position the temperature control valve accordingly.

Safety Evaluation Summary

It was concluded from the safety evaluation that the plant service water system is a non-safety related, Quality Class 2, system and the replacement of the temperature transmitter and temperature controller with a single component will have no impact on the operation of the plant. This is considered an equivalent change which does not have the potential to affect any previously evaluated accident or transient. The main generator hydrogen temperature control system performs no safety function.

- **Technical Evaluation Request 97-0136** (SE 97-130)

This technical evaluation request provided for the replacement of demineralized water system level control valve DW-LCV-11 and associated air operator. This is an equivalent change and the existing gate pattern valve was replaced with a globe pattern valve and a new compatible air operator that will use the existing control air supply.

Safety Evaluation Summary

It was concluded from the safety evaluation that the equivalent replacement components will conform to the applicable design specification requirements for this non-ASME, Quality Class 2, Seismic Class 2, non-safety related application.

Implementation and testing of the replacement components during any operational mode does not have the potential to affect any previously analyzed accident or malfunction, will not create the possibility of a different type of accident or equipment malfunction, and will not affect the Technical Specifications in any manner.

- **Work Order FZJ5** (SE 97-104)

This work order provided for the installation of a new WNP-2 site building, removal of various existing temporary trailers, and a revision to the overall site plan figure, which is also FSAR Figure 1.2-1.

Safety Evaluation Summary

It was concluded from the safety evaluation that installation of a new building and removal of various existing temporary trailers will not affect any important to safety structures, systems or components. The area is far removed from the power block and any system associated with the power block.

The external walls of power block structures provide adequate protection against adverse interactions between the new structure and any important to safety system or component within the power block. Tornado-generated missile criteria bound any potential impacts between the new building structure and important to safety structures, systems or components within the protected area. The actual location of the new building with respect to the power block and important to safety structures, systems or components precludes any interactions between these structures and components due to a design basis earthquake.

2.7 Regulatory Commitment Changes

This section contains information pertaining to Regulatory Commitment Changes (RCC) and is included pursuant to the NEI Guideline for Managing NRC Commitments. Included are those commitment changes that satisfied the NEI criteria for reporting.

This section does not include those commitment changes where the Staff was notified of the change under separate correspondence.

- **RCC-120947-00** (High High Rad and Very High Rad Areas)

  The original commitment description is, "The requirements for control and verification of locking doors to High High Rad and Very High Rad areas have been strengthened to include a physical check of the locked condition of the padlocks as the prescribed method for verifying the locked status of such doors . . . In addition, the verification function has been assigned to the Health Physics (HP) technicians." The commitment was made in a revised response to NRC Inspection Report 95-30. [Reference Letter GO2-96-035, dated February 22, 1996, JV Parrish (SS) to NRC, "NRC Inspection Report 95-30, Revision to Reply to Notice of Violation."]

  This commitment was revised. The revised commitment only deletes the last sentence of the original commitment where the verification function has been assigned to the HP technicians. The basis for revision is that Radiation Protection management has determined that the hands-on second verification function may be performed adequately by any other individual, not just by HP technicians. By also allowing an individual (other than Radiation Protection personnel) to perform the second verification, the second verification can be performed in a more timely manner and in most cases upon exit.

  Actual locking and initial verification will continue to be performed by Radiation Protection personnel.

- **RCC-144701-00** (Required Reading)

  The original commitment description is, ". . . only those Operating Experience Review (OER) related items determined critical for classroom training, via the Training Update System process will be incorporated into the scheduled requalification cycle. All other OER information will be addressed by the licensed operators in required reading administered by the Operations department." The commitment was made in response to NRC Examination Report OL-09-01. [Reference Letter GO2-93-045, dated February 25, 1993, GC Sorensen (SS) to JB Martin (NRC), "WNP-2 Licensed Operator Requalification Training Program Corrective Action Plan."]

This commitment was revised to, "In place of required reading, information will be conveyed by crew and staff meetings. If the items are part of corrective actions that have to be documented, a training attendance record will be included to document attendance. Operating experience reviews are also addressed using the Problem Evaluation Request process. During the Problem Evaluation Request resolution process, corrective action plans are developed that would address any training issues. Corrective action plans are tracked to completion by means of the Plant Tracking Log."

The revision does not alter the commitment to have in place a method to assure each licensed reactor operator and senior reactor operator is cognizant of OER issues.

- **RCC-144697-00** (Required Reading)

The original commitment description is, "Emergency Procedures Training following minor revisions is controlled by in-place administrative procedure wherein operator sign-off is required for the designated reading assignment. Major revisions will be addressed by required reading assignments and one of the following methods, depending on the magnitude of the changes: classroom training, control room walkthrough of simulator exercises." The commitment was made in response to an NRC request for additional information pertaining to initial procedural development. [Reference Letter GO2-83-0675, dated July 29, 1983, GC Sorensen (SS) to A Schwencer (NRC), "Procedures Generation Package for WNP-2, Request for Additional Information."]

This commitment was revised to, "In the place of required reading, information will be conveyed by crew and staff meetings. If items are part of corrective actions that have to be documented, a training attendance record will be included to document attendance."

The commitment revision will not alter the commitment to have in-place a method to assure each licensed operator and senior operator is cognizant of changes made to procedures.

- **RCC-106318-00** (Vendor Contact)

The original commitment description is, "Annual contact will be attempted with the vendor of each key system and component with issuance of a form letter being the primary method of contact." This commitment was made in response to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," and Generic Letter 90-03, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2, Part 2, Vendor Interface for Safety-Related Components." [Reference Letter GO2-90-161, dated September 28, 1990, GC Sorensen (SS) to NRC, "Amended Response to Generic Letter 83-28 (Item 2.2.2) and Response to Generic Letter 90-03 Regarding Vendor Interface for Safety Related Components (TAC 60758 and 76314)."]

This commitment was revised to state that the vendor re-contact will be on a periodic basis. This revision is necessary to support the current proposal to utilize a separate vendor, as part of the United Services Alliance program, to perform the vendor re-contacts. The selected vendor has established a three year cycle to contact the equipment supply vendors.

Generic Letter 90-03 required that licensees establish a periodic re-contact program with vendors of certain equipment to obtain updated or revised technical documents supplied by the vendor, concerning the piece of plant equipment supplied.

The commitment revision maintains compliance with Generic Letter 90-03, which requires "periodic" and not "annual" contact with equipment supply vendors.

- **RCC-104409-00** (ESF System Alignment)

The original commitment description is, ". . . all ESF systems are aligned and are operated as outlined in their respective Volume 2 Plant Procedures Manual. These system alignments are initially independently verified. Our program also requires independent verification following each major outage." The commitment was made in response to NUREG 0737, "Clarification of TMI Action Plan Requirements." [Reference Letter GO2-83-1044, dated November 11, 1983, GC Sorensen (SS) to A Schwencer (NRC), NUREG-0737, TMI-2 Action Plan, Item II.K.1.10 and II.K.1.5, Confirmatory Issue 22."]

This commitment was deleted. The basis for deletion is that refueling outages are divisionally separated.. Some systems will be undisturbed and will remain operable the entire outage. No regulatory requirement was identified that would require independent system verification of valve position of systems that had no maintenance or testing performed on them.

Plant Procedure 1.3.1, "Operating Policies, Programs and Practices," contains sufficient guidance for performing alignment and independent verification of required systems. There are other programs in place that prevent valves and breakers from becoming mispositioned. Deleting verification of these systems that have not been disturbed will also save accumulated dose.

- **RCC-106079-00** (Foreign Material in Control Panels)

The original commitment description is, "Scheduled inspections (once per quarter per the Scheduled Maintenance System) of panels will be made." The commitment was made in response to NRC Inspection Report 86-31. [Reference Letter GO2-86-960, dated October 20, 1986, GC Sorensen (SS) to JB Martin (NRC), "NRC Inspection Report 86-31, Reply to Notice of Violation."]

This commitment was deleted. At the time of the 1986 Notice of Violation, it was reported that, "A complete and documented survey was performed of all safety-related panels throughout the Plant and in the auxiliary buildings. Material found loose in the Control Room panels and other panels throughout the Plant have been removed and placed in proper storage."

The foreign material discovered in the panels was mostly debris and equipment left from construction and start-up time-frame. Management expectations about removing materials after a job is complete have been substantially strengthened. A review of the results of the scheduled maintenance system inspection for the past two years disclosed that there have been no cases of foreign material found in safety-related panels in the main control room.

This inspection is no longer necessary. The expectation to remove debris from panels and other areas after work completion has been set by management and is reinforced by plant procedures. This is a standing expectation of all personnel who perform work in the plant.

- RCC-85735-00 (POC Member Performance)

The original commitment description is, "The Plant Manager will develop performance expectations for members of POC. The Plant Manager will then provide quarterly feedback to members and the members' management on how well these expectations are being met." This commitment was made in response to NRC Systematic Assessment of Licensee Performance (SALP) Report 93-04. [Reference Letter GO2-93-125, dated May 27, 1993, WG Counsil (SS) to BH Faulkenberry (NRC), "Response to the 1993 Systematic Assessment of Licensee Performance (SALP)."]

This commitment was deleted. At the time this commitment was made, the Plant General Manager was not in frequent attendance at Plant Operations Committee (POC) meetings. Since March 1996, the Plant General Manager has attended POC regularly. Feedback on personnel performance (as POC members) is provided on a real-time basis as needed. Quarterly feedback is deemed unnecessary based upon current practice. In addition, POC meeting critiques are conducted at the conclusion of each POC meeting.

- RCC-137628-00 (CPR limit)

The original commitment description is, "Procedures were revised to require CPR greater than 2.2 between 25% power and 50% core flow." The commitment was made in LER 92-37-00. [Reference Letter GO2-97-218, dated September 14, 1992, JW Baker (SS) to NRC, "Licensee Event Report No. 92-037 - Manual Reactor Scram due to Core Instability."][5]

This commitment has been deleted. The critical power ratio limit has been deleted (removed from the applicable procedures). The three options from the BWR Owners Group (BWROG) that continue to be implemented are:

- On-line core stability monitor [Advanced Nuclear Noise Analysis (ANNA) Monitoring].
- Pre-analysis of reactor state point conditions for stability.
- Monitoring of control room neutron instruments, with the requirement of a manual scram upon indications of a reactor-instability-induced power oscillation (limits and precautions of startup and shutdown procedures).

In LER 92-037-03 it was stated that, "Supply System actions to address BWROG recommended actions may result in modification or suspension of the actions described in the enclosed Licensee Event Report . . . and no future revisions to the enclosed LER are planned."

The BWROG guidelines recommend that one of five options be adapted by each utility.[6] Three of the five options are implemented at WNP-2. Even with the deletion of the critical power ratio limit, implementation of the three remaining options provides sufficient defense in depth to prevent recurrence of core instability.

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5 This commitment was contained in the original LER and all supplemental reports (i.e., LER 92-037-00, LER 92-037-01, LER 92-037-02 and LER 92-037-03.

6 BWROG-94079, "BWR Owners' Group Improved Guidelines for Stability Interim Corrective Actions," dated June 6, 1994

- **RCC-141644-00** (Slings)

The original commitment description pertained to the use of slings at WNP-2. As the result of a request for additional information by EG&G Idaho for evaluation of WNP-2 compliance to NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants," the Supply System provided a detailed listing of all applicable slings, including rated capacity, heavy loads used with, and cranes used on (if applicable), and indication of compliance with the criteria of ANSI B30.9-1971, "Slings," as amended by NUREG 0612, Section 5.1.1(5) concerning dynamic loads and special marking for each sling. The Supply System response included a table providing the information requested. In addition, the Supply System response provided information for the size and material that was being used at that time or was intended to be used to meet ANSI B30.9 and NUREG 0612 requirements.

This commitment has been revised. The commitment revision is to remove the specificity of the sling description and allow use of equivalent or better slings that meet the base requirements of NUREG 0612 and ANSI B30.9. The commitment has been revised to read, "The slings listed are representative of those that comply with Table 7 of ANSI B30.9 . . . other slings that meet or exceed the requirements of ANSI B30.9 and NUREG 0612 may be used."

The specific details of the slings to be used are not needed. It was not intended to restrict use of only slings of specific size or material. The original information was provided to show that what was being used would meet the requirements.

Because the slings and the associated lifts fall under the Heavy Loads Program (NUREG 0612 and ANSI B30.9-1971), all associated requirements are met. Specifying the actual size and material of the slings was unduly restrictive and provided no added safety or benefit. In fact, it would prevent using larger or higher rated slings in specific situations.

- **RCC-105572-00** (Penetration Seals)

The original commitment description is, "Engineering will provide additional re-sealing instructions on the work implementation summary sheet of those future Plant Modifications." The commitment was made in LER 87-29. [Reference Licensee Event Report 87-029, dated October 16, 1987, CM Powers (SS) to NRC, "Plant Technical Specification Fire-Rated Floor Penetration Impaired During Plant Design Modification - Inadequate Training."]

This commitment has been deleted. The commitment did not provide an effective barrier and the other barriers incorporated as a result of the LER were effective in identifying the need for seals in the original design process. The plant modification process identifies the

need for penetration seals and is considered adequate. The basic design change process includes an installation instruction summary sheet and the plant modification includes a plant modification package checklist.

There is no "work implementation summary sheet" described by either process. The basic design change review process requires that a fire protection engineer sign off on all design change packages.

This fire protection engineer review has proven to be the most effective barrier to ensure that penetration seals are identified in the initial issue of the basic design change.

APPENDIX
WNP-2
ANNUAL OPERATING REPORT
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

10 CFR PART 20

Facility: 02

Report for Calendar Year: 1997

This report was produced with direct reading dosimeter data

Number of Persons Receiving Over 100 millirem is 591 Total MAN-REM: 232.36

		Number of Individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contracto and Other
OPERATIONS AND SURVEILLANCE	Maintenance Personnel	59.59	2.71	36.43	9.945	0.405	4.483
	Operating Personnel	28.82	0.04	0.87	9.740	0.001	0.187
	Health Physics Personnel	6.03	0	0.71	0.988	0.000	0.102
	Supervisory Personnel	4.71	0.16	0.59	0.453	0.021	0.138
	Engineering Personnel	7.89	4.92	4.88	1.289	0.510	0.857
ROUTINE MAINTENANCE	Maintenance Personnel	77.88	1.86	128.36	38.757	1.009	51.234
	Operating Personnel	1.86	0.00	0.12	4.114	0.000	0.201
	Health Physics Personnel	22.20	0.00	32.44	8.705	0.000	10.045
	Supervisory Personnel	1.89	0.38	3.03	1.622	0.105	1.034
	Engineering Personnel	7.12	4.69	4.61	2.666	0.967	1.204
INSERVICE INSPECTION	Maintenance Personnel	4.71	0.09	29.99	5.345	0.340	23.902
	Operating Personnel	0.46	0.00	0.04	1.323	0.000	0.062
	Health Physics Personnel	0.35	0.00	0.37	0.516	0.000	0.690
	Supervisory Personnel	0.28	0.00	1.25	0.347	0.000	0.456
	Engineering Personnel	0.75	0.22	5.06	0.538	0.259	1.841
SPECIAL MAINTENANCE	Maintenance Personnel						
	Operating Personnel						
	Health Physics Personnel			(See attached sheets)			
	Supervisory Personnel						
	Engineering Personnel						
WASTE PROCESSING	Maintenance Personnel	2.35	0.37	0.49	0.459	0.556	0.057
	Operating Personnel	0.00	0.00	0.73	0.000	0.000	0.166
	Health Physics Personnel	0.17	0.00	0.21	0.183	0.000	0.161
	Supervisory Personnel	0.10	0.01	0.00	0.009	0.001	0.000
	Engineering Personnel	0.24	0.16	0.00	0.039	0.014	0.000
REFUELING	Maintenance Personnel	21.10	0.09	10.84	16.490	0.011	2.764
	Operating Personnel	0.23	0	0.00	0.319	0.000	0.000
	Health Physics Personnel	1.73	0	2.09	0.802	0.000	0.623
	Supervisory Personnel	4.19	0.45	0.00	1.713	0.127	0.000
	Engineering Personnel	0.40	10.41	8.97	0.021	1.770	1.602
TOTAL	Maintenance Personnel	170.14	5.20	234.04	73.226	2.468	95.986
	Operating Personnel	31.37	0.04	1.76	15.496	0.001	0.616
	Health Physics Personnel	31.02	0.00	36.00	12.238	0.000	11.725
	Supervisory Personnel	11.33	1.00	4.99	4.410	0.254	1.660
	Engineering Personnel	19.05	20.94	23.91	5.048	3.601	5.570
Grand Total		262.91	27.18	300.70	110.418	6.324	115.557

10 CFR PART 20

Facility: 02

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APPENDIX
WNP-2
ANNUAL OPERATING REPORT
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Report for Calendar Year: 1997

		Number of Individuals			Year to Date Dose (MAN-REM)		
SPECIAL MAINTENANCE		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contracto and Othe
1. Layup/Rebuild Control Rod Drive Mechanisms	Maintenance Personnel	0.00	0.04	10.61	0.000	0.067	5.144
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.28	0.00	0.08	0.546	0.000	0.045
	Supervisory Personnel	0.03	0.00	0.05	0.055	0.000	0.014
	Engineering Personnel	0.00	0.25	0.18	0.000	0.037	0.030
2. Rebuild/rework Penetration Seals	Maintenance Personnel	0.00	0.02	7.58	0.000	0.046	3.677
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.06	0.000	0.000	0.034
	Supervisory Personnel	0.02	0.00	0.04	0.035	0.000	0.010
	Engineering Personnel	0.95	0.17	0.12	0.177	0.025	0.020
3. Replace Reactor Water Cleanup Valves	Maintenance Personnel	0.00	0.01	3.78	0.000	0.023	1.831
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.10	0.00	0.03	0.183	0.000	0.016
	Supervisory Personnel	0.02	0.00	0.02	0.025	0.000	0.005
	Engineering Personnel	0.00	0.09	0.06	0.000	0.013	0.010
4. Paint/Label 522' Level, Reactor Bldg	Maintenance Personnel	1.02	0.00	0.00	0.505	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	0.01	0.00	0.00	0.015	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
5. Shield FPC Heat Exchangers	Maintenance Personnel	0.51	0.00	0.29	0.253	0.000	0.143
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.04	0.00	0.00	0.074	0.000	0.000
	Supervisory Personnel	0.01	0.00	0.00	0.011	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
6. Replace Reactor Recirculation Valve 19 & Modify Support	Maintenance Personnel	0.34	0.01	1.53	0.169	0.011	0.741
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.00	0.00	0.02	0.000	0.000	0.009
	Supervisory Personnel	0.01	0.00	0.01	0.015	0.000	0.003
	Engineering Personnel	1.11	0.04	0.04	0.208	0.006	0.006
7. Replace Bonnet, Reactor Core Isolation Cooling Valve 76	Maintenance Personnel	0.00	0.00	0.84	0.000	0.000	0.408
	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	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
8. Replace Internal Radiation Monitor Detector 2H	Maintenance Personnel	0.78	0.00	0.00	0.384	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	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000

10 CFR PART 20

Facility: 02

APPENDIX
WNP-2
ANNUAL OPERATING REPORT
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Report for Calendar Year: 1997

This report was produced with direct reading dosimeter data

		Number of Individuals			Year to Date Dose (MAN-REM)		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contracto and Othe
SPECIAL MAINTENANCE							
9. Install Scaffolding in Wetwell	Maintenance Personnel	0.00	0.00	0.81	0.000	0.000	0.395
	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	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
10. Replace PSR Valve X77A/3	Maintenance Personnel	0.33	0.00	1.09	0.164	0.000	0.529
	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	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.49	0.00	0.00	0.092	0.000	0.000
11. Replace Connectors on LPRM Detectors	Maintenance Personnel	0.56	0.00	0.00	0.276	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	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
12. Setup and tear down R-12 Health Physics Equipment	Maintenance Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.09	0.00	0.00	0.169	0.000	0.000
	Supervisory Personnel	0.01	0.00	0.00	0.010	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
13. Replace Relay on Tip-Drive-1C	Maintenance Personnel	0.15	0.00	0.00	0.074	0.000	0.000
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.02	0.00	0.00	0.040	0.000	0.000
	Supervisory Personnel	0.00	0.00	0.00	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
14. Rotate Reactor Recirculation Valve 14B	Maintenance Personnel	0.00	0.00	0.23	0.000	0.000	0.113
	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
15. Replace Temperature Element, SPTM-TE-2A	Maintenance Personnel	0.20	0.00	0.00	0.099	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
16. Clean Drain, 522' Pump Room A	Maintenance Personnel	0.00	0.00	0.19	0.000	0.000	0.090
	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000

10 CFR PART 20

Facility: 02

APPENDIX
WNP-2
ANNUAL OPERATING REPORT
RADIATION EXPOSURE RECORDS
WORK AND JOB FUNCTION REPORT

Report for Calendar Year: 1997

This report was produced with direct reading dosimeter data

		----Number of Individuals---			-----Year to Date Dose----- (MAN-REM)		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contracto and Othe
SPECIAL MAINTENANCE							
17. Modify Condensate Demins 1B Septa	Maintenance Personnel	0.18	0.00	0.00	0.087	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
18. Replace Jumper Cables SRM Detectors	Maintenance Personnel	0.31	0.00	0.00	0.154	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
19. Install Thermo-LAG on Terminal B	Maintenance Personnel	0.00	0.00	0.14	0.000	0.000	0.066
	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	0.005	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
20. Miscellaneous	Maintenance Personnel	0.14	0.00	0.84	0.067	0.000	0.409
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.02	0.00	0.00	0.031	0.000	0.000
	Supervisory Personnel	0.01	0.00	0.00	0.015	0.000	0.000
	Engineering Personnel	0.09	0.00	0.00	0.018	0.000	0.000
Special Maintenance Totals	Maintenance Personnel	4.51	0.08	27.93	2.231	0.147	13.546
	Operating Personnel	0.00	0.00	0.00	0.000	0.000	0.000
	Health Physics Personnel	0.54	0.00	0.18	1.044	0.000	0.104
	Supervisory Personnel	0.16	0.00	0.12	0.266	0.000	0.032
	Engineering Personnel	2.65	0.54	0.39	0.495	0.081	0.066