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February 25, 1999
GO2-99-041

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 1998

The annual operating report for calendar year 1998 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,

D.K. Atkinson for

RL Webring
Vice President, Operations Support/PIO
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1/1

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Attachment

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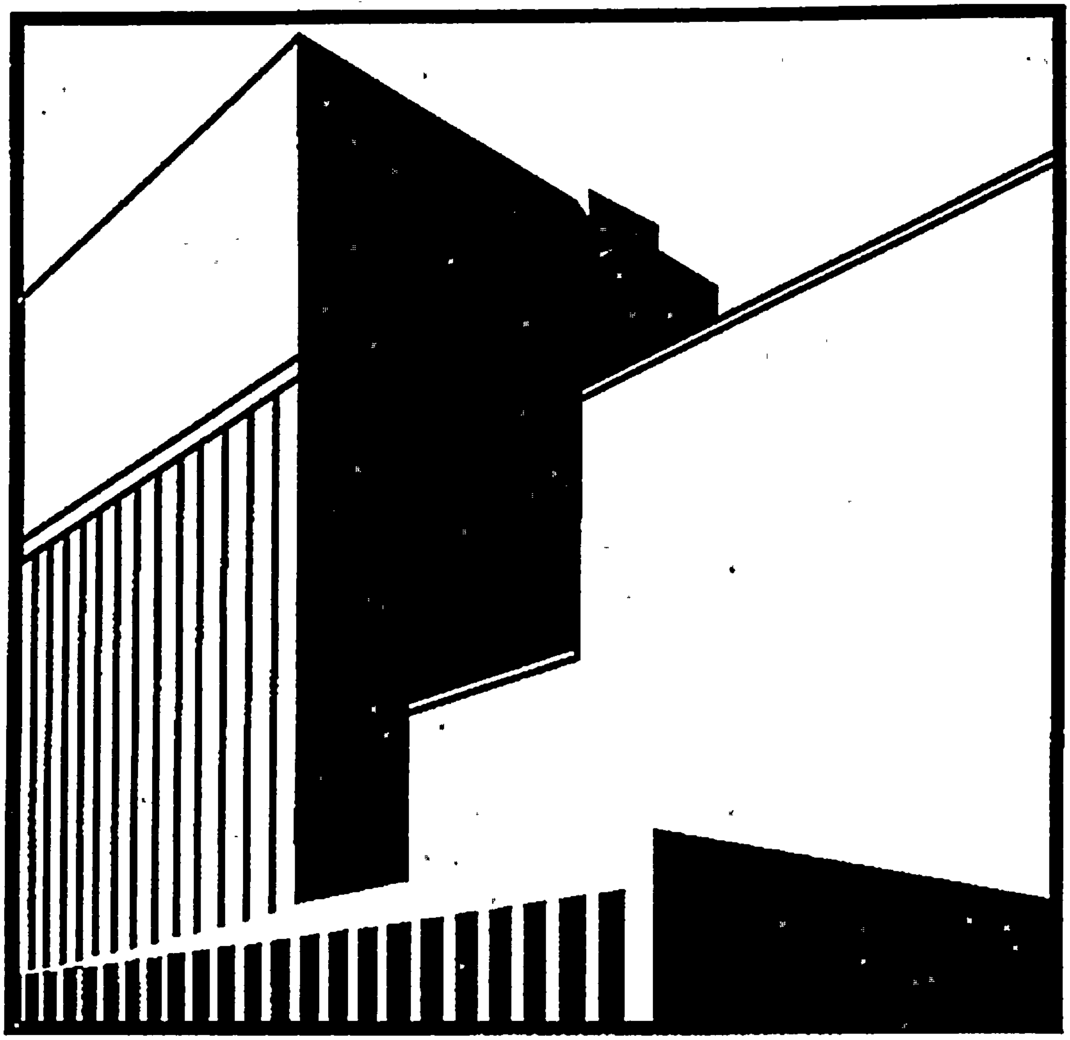

WNP-2 1998
ANNUAL OPERATING REPORT

WASHINGTON PUBLIC POWER
SUPPLY SYSTEM

9903100229

WASHINGTON NUCLEAR PLANT NO. 2

ANNUAL OPERATING REPORT

1998

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

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

WNP-2 1998
Annual Operating Report

Table of Contents

1.0 Introduction

The 1998 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 11, 1998 the plant was in a third-best operational run of 243 days when the closure of an inboard main steam system isolation valve initiated an automatic reactor trip. The reason for the closure of the isolation valve was loss of pneumatic actuating supply pressure due to cyclic fatigue failure of the supply line tubing. The tubing was re-worked to a configuration less susceptible to cyclic fatigue. Inspections and stress evaluations were also performed on the remaining main steam isolation valve supply tubing configurations. In addition, all fittings were checked for leaks.

Following recovery efforts, the plant was re-started on March 18, 1998. On March 21, 1998 the plant was returned to 100 percent full power operation and remained at or near that level until April 18, 1998 when the unit was shutdown for the annual maintenance and refueling outage.

The annual maintenance and refueling outage officially ended on June 13, 1998 with final synchronization to the electrical grid. On June 15, 1998, with the plant at 37 percent power and in power ascension following the annual maintenance and refueling outage, plant operators commenced a controlled reactor shutdown as required by the Technical Specifications because the cabling for a traversing incore probe drive machine would not retract. The undervessel tubing for the affected traversing incore probe channel was found uncoupled. Accordingly, the affected tubing was re-attached and tightened and other traversing incore probe system tubing was inspected.

On June 17, 1998, with the plant in a shutdown condition, a significant water hammer event in plant fire protection system piping resulted in the rupture of a fire protection system valve located in the reactor building northeast stairwell. Water from the ruptured fire protection valve flooded the stairwell, a residual heat removal system pump room, and the low pressure core spray system pump room.

After verifying that no fire or threat of fire existed, Operations personnel shut off the operating fire pumps and terminated the source of the flooding. The reason for the event was determined to be inadequate design of the fire protection system. Extensive flooding damage assessment and repair efforts were performed and a detailed review of the design of the fire protection system was conducted.

Following extensive flood recovery efforts, the plant was restarted and synchronized to the electrical grid on July 4, 1998. On July 8, 1998 the plant returned to 100 percent full power operation. On August 4, 1998, reactive load oscillations were observed during performance of the emergency diesel generator E-DG-2 monthly operability test. Accordingly, emergency diesel generator E-DG-2 was shutdown and declared inoperable in accordance with the Technical Specifications.

Subsequent troubleshooting and repair efforts associated with the emergency diesel generator voltage regulation system failed to restore E-DG-2 to an operable condition within the time-frame allowed by the Technical Specifications. Accordingly, on August 7, 1998 a plant shutdown was initiated and completed as required by the Technical Specifications. During follow-up testing of emergency diesel generator E-DG-2, the output breaker failed to automatically close. Following extensive trouble shooting efforts and replacement of the E-DG-2 static exciter voltage regulator, a design change was implemented which replaced the existing 24 VDC and 125 VDC power supplies in the governor controls for emergency diesel generators E-DG-1 and E-DG-2.

During the forced outage, battery service testing was performed on the Division 1, 2 and 3 125 VDC batteries and the Division 1 250 VDC battery. This was performed in response to a July 15, 1998 event where it was discovered that Technical Specification Surveillance Requirement 3.8.4.7 had not been fulfilled within the specified frequency, plus the allowed extension time, for Division 2, 125 VDC, battery E-B1-2. Following receipt of discretionary enforcement, a follow-up exigent Technical Specification amendment request was submitted and approved to allow for the performance discharge test described in Surveillance Requirement 3.8.4.8 to be performed in lieu of the service test for battery E-B1-2 until Surveillance Requirement 3.8.4.7 could be performed during the next refueling outage, or a forced outage of sufficient duration.

During the forced outage, repairs were made to a socket weld on a three-quarter-inch reactor recirculation system sensing line that had developed a leak during operation. An increasing trend of unidentifiable drywell leakage (0.56 gpm) had been observed and closely monitored by plant personnel following restart from the recent maintenance and refueling outage. The instrument line had been included in the walkdowns associated with startup from the refueling outage and no leak was visible at that time.

On August 22, 1998 following successful forced outage recovery efforts, the plant was synchronized to the electrical grid. The plant was returned to 100 percent full power operation on August 23, 1998 and remained on line for the rest of the year.

Refueling Outage Summary

The R-13 maintenance and refueling outage was successfully completed during 1998. Significant planned and emergent activities included:

- Replacement of ten emergency core cooling system suction strainers in the suppression chamber.
- Replacement of 132 fuel assemblies.
- Inspection of safe-end welds.
- Inspection of core shroud welds.
- Refurbishment of ten main steam relief valves.
- Replacement of nine local power range monitor strings.
- Replacement of ten carbon steel control rod drive system accumulator tanks with stainless steel tanks.
- Testing of 37 snubbers located in the drywell and reactor and turbine buildings.

Other Highlights

During 1998, new power generation records were set for the months of February and October. The net electrical generation was 753,683 megawatt-hours and 856,655 megawatt-hours respectively.

Significant efforts were undertaken in response to the unexpected flooding event due to the rupture of a fire protection system valve. Operators took timely and decisive actions to isolate the flood source, activate the emergency response organization, and notify the appropriate authorities. The organization responded well to the challenges of the event and demonstrated effective teamwork with a focus on safety.

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 1998

- At the beginning of the month, the plant was at 100 percent full power operation. On January 7, 1998 power was reduced to 76 percent for turbine valve testing and control rod adjustments.

- On January 8, 1998 the plant was returned to 100 percent full power operation and remained at or near that level for the remainder of the month.

February 1998

- At the beginning of the month, the plant was at 100 percent full power operation. On February 3, 1998, during the performance of emergency diesel generator surveillance testing, a control room operator mistakenly tripped the supply breaker for 4.16 kV electrical bus SM-2 by inadvertent operation of the breaker handswitch. This resulted in the loss of electrical bus SM-2 and the subsequent tripping of condensate system pump COND-P-1B, condensate system booster pump COND-P-2B, circulating water system pump CW-P-1B and the supply breaker to bus SM-4.

  This also resulted in the automatic starting of high pressure core spray system diesel generator HPCS-GEN-DG-3. Plant control room operators took prompt corrective action to lower total core flow and successfully stabilize the plant at approximately 75 percent power. Operations supervision conducted appropriate counseling to address the human performance errors associated with this event and a station-wide stand down was conducted to review this event and other issues (Reference LER 98-001-00).

- On February 4, 1998, following recovery efforts, the plant was returned to 100 percent full power operation and remained at or near that level for the remainder of the month.

March 1998

- The plant entered the month at 100 percent full power operation. On March 1, 1998 power was reduced to 60 percent to investigate a suspected main condenser tube leak that was causing perturbations to primary coolant chemistry parameters. The apparent leak was pinpointed to the main condenser "C" water box and that section of the condenser was isolated and drained for inspection. However, during the inspection effort, no leak was found and the decision was made to return to full power operation and closely monitor water chemistry.

- On March 4, 1998 the plant was returned to 100 percent full power operation. On March 8, 1998 power was reduced to approximately 90 percent for a control rod pattern adjustment and was returned to 100 percent full power operation the same day.

- On March 11, 1998, following a third-best operational run of 243.5 days, WNP-2 experienced a plant transient and automatic reactor trip that was initiated by the closure of inboard main steam system isolation valve MS-V-2D. Immediately after the closure of the isolation valve, a reactor trip occurred due to instantaneous high neutron flux, and steam flow was diverted to the remaining three steam lines, causing a high steam flow condition of 140 percent flow and the automatic closure of all main steam isolation valves.

  During the event, reactor pressure immediately peaked and was limited to approximately 1090 psig by the automatic lifting of two main steam safety relief valves. Due to the pressure peak, sensed vessel level decreased to Level-2 (-50 inches) which caused initiation of the reactor core isolation cooling and high pressure core spray systems. Due to the loss of drywell cooling, drywell pressure peaked at about 1.60 psig approximately 11 minutes into the event. The drywell pressure peak caused the automatic starting of the emergency diesel generators and partial makeup of the low pressure emergency core cooling system start logic (Reference LER 98-002-00).

  At approximately 50 minutes into the event, with all control rods fully inserted into the reactor core, a second reactor scram signal occurred due to a reactor pressure vessel low level condition at Level-3 (+13 inches). This was due to a less than optimal post-trip operational strategy for resetting scram signals in these conditions. Accordingly, changes in post-trip operational strategy were implemented in plant procedures (Reference LER 98-003-00).

  The reason for the closure of isolation valve MS-V-22D was loss of pneumatic actuating supply pressure due to cyclic fatigue failure of the supply line tubing. The tubing was re-worked to a configuration less susceptible to cyclic fatigue. Inspections and stress evaluations were also performed on the remaining main steam isolation valve supply tubing configurations. In addition, all fittings were checked for leaks.

- On March 18, 1998 following recovery efforts, the plant was re-started. On March 21, 1998 the plant was returned to 100 percent full power operation and remained at or near that level for the remainder of the month.

April 1998

- At the beginning of the month, the plant was at 100 percent full power operation and the unit began an end-of-fuel cycle power coastdown. A final feedwater temperature reduction operational strategy was also implemented.

- On April 17, 1998 the plant was in the second stage of final feedwater temperature reduction and had coasted down to 98 percent when power was reduced in preparation for the annual maintenance and refueling outage.

- On April 18, 1998 the main turbine generator was removed from service and the plant was shutdown for the annual maintenance and refueling outage.

May 1998

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

June 1998

- The annual maintenance and refueling outage officially ended on June 13, 1998 with final synchronization to the electrical grid, and the plant was in power ascension following the annual maintenance and refueling outage.

- On June 15, 1998, with the plant at 37 percent power, plant operators commenced a controlled reactor shutdown as required by the Technical Specifications due to a traversing incore probe system malfunction. The cabling for a traversing incore probe drive machine would not retract and plant personnel were unable to close the associated penetration isolation ball valve due to the inability to withdraw the detector cabling. Control room operators took appropriate and timely action to maneuver the plant to a shutdown condition. On June 15, 1998, at 1947 hours, the plant entered cold shutdown.

  The cause of the event was improper installation of the traversing incore probe tubing due to inadequate self-checking. It was determined that the tubing had been installed in the reverse direction and not adequately tightened during tube removal and replacement efforts in the recently-completed maintenance and refueling outage. This resulted in the tube fittings becoming loose.

  Corrective actions included re-attaching the affected tubing, inspecting other traversing incore probe tubing and discussing this event and the importance of self-checking with plant personnel (Reference LER 98-010-00).

- On June 17, 1998, with the plant in a shutdown condition, a significant water hammer event in plant fire protection system piping resulted in the rupture of fire protection system valve FP-V-29D located in the reactor building northeast stairwell.

  Subsequent investigation determined that the water hammer was due to creation of a void in the vertical fire protection riser when smoke from maintenance activities in a diesel generator room activated a detector.

Water from the ruptured fire protection valve flooded the stairwell, a residual heat removal system pump room, and the low pressure core spray system pump room. The resulting flood water completely submerged the residual heat removal system pump and motor and the associated keep-fill pump. The level in the low pressure core spray system pump room rose to just below the pump motor and completely submerged the associated keep-fill pump. After verifying that no fire or threat of fire existed, Operations personnel shut off the operating fire pumps and terminated the source of flooding.

The reason for event was determined to be inadequate design of the fire protection system. Extensive flooding damage assessment and repair efforts were performed and a detailed review of the fire protection system design was conducted (Reference LER 98-011-00).

- The plant remained in a shutdown condition for the remainder of the month.

July 1998

- At the beginning of the month the plant was in a shutdown condition due to the forced outage from the fire protection system flooding event.
- Following extensive flood recovery efforts, synchronization to the electrical grid occurred on July 4, 1998. On July 8, 1998 the plant returned to 100 percent full power operation.
- On July 10, 1998 power was reduced to approximately 85 percent for a control rod set and testing. The plant was returned to 100 percent full power operation the same day and remained at or near that level for the remainder of the month.

August 1998

- At the beginning of the month, the plant was at 100 percent full power operation. On August 4, 1998, reactive load oscillations were observed during performance of the emergency diesel generator E-DG-2 monthly operability test. Accordingly, emergency diesel generator E-DG-2 was shutdown and declared inoperable in accordance with the Technical Specifications. (It was later learned that the voltage regulation system for E-DG-2 had degraded.)

  On August 5, 1998 during troubleshooting efforts associated with the E-DG-2 apparent voltage regulator problems, vital electrical bus SM-8 and its associated loads were inadvertently deenergized causing several engineered safety feature system isolations and half-isolations to occur. During this event, reactor feedwater system pump RFW-P-1B tripped. By design, the tripping of pump RFW-P-1B resulted in an automatic reactor recirculation system runback and a power reduction to approximately 65 percent. The pump was returned to service and power was restored to approximately 70 percent.

The cause of the event was determined to be inadequate direction in the troubleshooting plan to respond to anticipated abnormal system responses. The trip of pump RFW-P-1B was initiated by relay interaction in the reactor feedwater turbine electrical overspeed trip circuit (Reference LER 98-013-00).

- Subsequent troubleshooting and repair efforts associated with the emergency diesel generator voltage regulation system failed to restore E-DG-2 to an operable condition within the time-frame allowed by the Technical Specifications. Accordingly, on August 7, 1998 a plant shutdown was initiated and completed as required by the Technical Specifications (Reference LER 98-014-00).

During follow-up testing of emergency diesel generator E-DG-2, the output breaker failed to automatically close. The testing was being performed to demonstrate operability following repair of the E-DG-2 static exciter voltage regulator. Following extensive trouble shooting efforts, it was determined that the power supply which feeds the speed switch electronics and the speed switch auxiliary relays in the generator control panel was undersized. Accordingly, a design change was implemented which replaced the existing 24 VDC and 125 VDC power supplies in the governor controls for emergency diesel generators E-DG-1 and E-DG-2.

During the forced outage, battery service testing was performed on the Division 1, 2 and 3 125 VDC batteries and the Division 1 250 VDC battery. This was performed in response to a July 15, 1998 event where it was discovered that Technical Specification Surveillance Requirement 3.8.4.7 had not been fulfilled within the specified frequency, plus the allowed extension time, for Division 2, 125 VDC, battery E-B1-2. Following receipt of discretionary enforcement, a follow-up exigent Technical Specification amendment request was submitted and approved to allow for the performance discharge test described in Surveillance Requirement 3.8.4.8 to be performed in lieu of the service test for battery E-B1-2 until Surveillance Requirement 3.8.4.7 could be performed during the next refueling outage, or a forced outage of sufficient duration. The reason for the noncompliance was that work practices used to convert those specific battery procedures were inadequate to ensure that the modified performance test discharge profile was incorporated during conversion to the Improved Technical Specifications (Reference LER 98-012-00 and LER 98-012-01).

During the forced outage, repairs were made to a socket weld on a three-quarter-inch reactor recirculation system sensing line that had developed a leak during operation. An increasing trend of unidentifiable drywell leakage (0.56 gpm) had been observed and closely monitored by plant personnel following restart from the recent maintenance and refueling outage. The instrument line had been included in the walkdowns associated with startup from the refueling outage and no leak was visible at that time. A metallurgical evaluation was performed which confirmed that the failure mechanism was attributed to fatigue failure and not stress corrosion cracking.

- On August 22, 1998 following successful forced outage recovery efforts the plant was synchronized to the electrical grid. The plant was returned to 100 percent full power operation on August 23, 1998.

- On August 24, 1998 power was reduced to approximately 87 percent for control rod pattern adjustments.

- On August 25, 1998 the plant was returned to 100 percent full power operation and remained at or near that level until August 31, 1998 when Operators began a downpower to approximately 85 percent for economic dispatch (load following) at the request of the Bonneville Power Administration.

September 1998

On September 1, 1998 the plant was returned to 100 percent full power operation and remained at or near that level for the remainder of the month.

October 1998

The plant entered the month at 100 percent power and operated at or near that level for the remainder of the month.

November 1998

- The plant entered the month at 100 percent power and operated at or near that level until November 7, 1998 when power was reduced to 56 percent for a control rod pattern exchange, turbine valve and bypass valve testing, data gathering on the main generator stator cooling water system, and minor modifications to the reactor feedwater drive control system.

- On November 8, 1998 power was raised to 65 percent and the plant was returned to 100 percent full power operation on November 10, 1998.

- On November 24, 1998 power was reduced to 80 percent for economic dispatch (load following) until November 27, 1998 at the request of the Bonneville Power Administration.

- On November 27, 1998 power was reduced to 65 percent for economic dispatch (load following) until November 30, 1998 at the request of the Bonneville Power Administration.

- On November 30, 1998 the plant was returned to 100 percent full power operation.

December 1998

- The plant entered the month at 100 percent power and operated at or near that level until December 16, 1998 when power was reduced to approximately 90 percent for a few hours to support a control rod pattern change and control rod maintenance. Power level was returned to 100 percent the same day.

- On December 20, 1998 power was reduced to 85 percent to support a control rod pattern change and turbine front standard testing. Power level was returned to 100 percent the same day.

- On December 28, 1998 power was reduced to 85 percent for economic dispatch (load following) until December 29, 1998 at the request of the Bonneville Power Administration.

- On December 29, 1998 power was reduced to 65 percent for economic dispatch (load following) and remained at that level for the rest of the month at the request of the Bonneville Power Administration.

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

New emergency core cooling system suction strainers were installed 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."

Inspections were conducted on several safe-end welds pursuant to commitments made in response to NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping."

A limited reactor core shroud weld inspection was performed of the four core shroud representative areas pursuant to commitments made in response to NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds."

Control Rod Drive System

Ten 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.

Control Room HVAC System

Control room emergency chilled water system heat exchanger CCH-CR-1B was opened and cleaned as part of an ongoing heater and heat exchanger tube cleaning and maintenance program. In addition, nondestructive examination (eddy current) was performed on the heat exchanger.

Emergency Diesel Generator System

The voltage regulator and associated components were replaced for emergency diesel generator E-DG-2 following an event where reactive load oscillations were observed during performance of the emergency diesel generator E-DG-2 monthly operability test.

The existing 24 VDC and 125 VDC power supplies in the governor controls for emergency diesel generators E-DG-1 and E-DG-2 were replaced following an event where reactive load oscillations were observed during performance of the emergency diesel generator E-DG-2 monthly operability test. It was subsequently determined that the power supplies which feed the speed switch electronics and the speed switch auxiliary relays in the generator control panels were undersized.

Emergency diesel generator system heat exchangers DCW-HX-A1 and DCW-HX-1A2 were opened and cleaned as part of an ongoing heater and heat exchanger tube cleaning and maintenance program. In addition, nondestructive examination (eddy current) was performed on both heat exchangers.

Low Pressure Core Spray System

The space heater contactor for low pressure core spray system pump motor LPCS-M-P/1 was cleaned, dried, and cycled mechanically as part of recovery efforts associated with the flooding event due to rupture of fire protection system valve FP-V-29D. The coil was also inspected and cleaned, resistance measurements were taken and fuses were replaced. As a precaution, the low pressure core spray system pump motor feeder cable was also megger tested.

Low pressure core spray system pump motor LPCS-M-P/2 was replaced as part of recovery efforts associated with the flooding event due to the rupture of fire protection system valve FP-V-29D. The motor feeder cable was also megger tested.

Low pressure core spray system flow control valve motor LPCS-M-FCV/11 was replaced as part of recovery efforts associated with the flooding event due to the rupture of fire protection system valve FP-V-29D. New fuses and an overload heater were installed. The motor-operator was overhauled and the torque and limit switches were inspected.

Main Steam System

The tubing associated with main steam system isolation valve MS-V-22D was re-worked to a configuration less susceptible to cyclic fatigue following an event where the valve closed, due to loss of pneumatic actuating supply pressure, because of cyclic fatigue failure of the supply line tubing. Inspections and stress evaluations were also performed on the remaining main steam isolation valve supply tubing configurations. In addition, all fittings were checked for leaks.

Ten safety relief valves were refurbished as part of an ongoing corrective maintenance program to minimize leakage into the wetwell. Leakage into the wetwell presents unnecessary operational challenges such as suppression pool temperature increases.

Nondestructive examination (eddy current) was performed on main steam system moisture separator/reheater heat exchangers MS-HX-1A-B2 and MS-HX-1B-D2 as part of an ongoing heater and heat exchanger tube cleaning and maintenance program.

Miscellaneous Systems

Numerous structures and components were evaluated and appropriately returned to service as part of recovery efforts associated with the flooding event due to the rupture of fire protection system valve FP-V-29D. These items included motors, relays, conduit, closed cable trays, transformers, pumps, snubbers, spring cans, insulation, valves, doors, seals, walls and instruments. The areas flooded were the reactor building northeast stairwell, the 422-foot elevation corridor, a residual heat removal system pump room and the low pressure core spray pump room.

Neutron Monitoring System

Nine local power range monitor strings were replaced as part of an ongoing long-life local power range monitor replacement program.

Residual Heat Removal System

Residual heat removal system pump motor RHR-M-P/2C was inspected, cleaned, dried and tested as part of recovery efforts associated with the flooding event due to the rupture of fire protection system valve FP-V-29D. The motor bearings were replaced and the feeder cables were megger tested. The motor was run-tested to verify proper current and voltages on all three phases under light load. Winding resistance and vibration measurements under light load conditions were also performed.

Residual heat removal system pump motor RHR-M-P/3 was inspected, steam cleaned and dried as part of recovery efforts associated with the flooding event due to the rupture of fire protection system valve FP-V-29D. A new bearing was installed and the motor feeder cable was megger tested.

Residual heat removal system heat exchanger RHR-HX-1A was opened and cleaned as part of an ongoing heater and heat exchanger tube cleaning and maintenance program. In addition, nondestructive examination (eddy current) was performed on the heat exchanger.

Traversing Incore Probe System

Traversing incore probe system drive machine cabling and detector were replaced. The associated tubing was repaired and other tubing was inspected following an event where it was discovered that tubing had been installed in the reverse direction and not adequately tightened during traversing incore probe tube removal and replacement efforts.

The traversing incore probe system "C" ball valve and actuator were replaced and the "B" indexer was rebuilt. Three explosive squib valve charges were also replaced.

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 and were 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. There was no evidence of failed fuel during the calendar year 1998 portion of Cycle 13. There was also no evidence of failed fuel for the calendar year 1998 portion of Cycle 14.

This input is provided solely for informational purposes and ease of reference. Regulatory Guide 1.16, Section C.1.b.(4), only requires reporting where there are indications of failed fuel.

2.5 10CFR50.46, Changes or Errors in ECCS LOCA Analysis Models

This section contains information relative to changes and errors in Emergency Core Cooling System (ECCS) cooling performance models.

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 non-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 impact 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 (SPC) fuel. This is the LOCA analysis of record for WNP-2.[1] The ABB methodology was used to license ABB SVEA-96 fuel.[2]

For 1998, there were no changes or errors in an ECCS LOCA analysis model or application for the SPC fuel.

For the ABB fuel, there also were no changes or errors in an ECCS LOCA analysis model for 1998. However, two changes were implemented in the application of the ABB ECCS evaluation methodology for the SVEA-96 reload fuel. The initial (pre-LOCA event) oxidation of the cladding was revised to reflect higher values based on in-plant measured data for assembly burnups greater than 30 MWd/kgU. The current WNP-2 SVEA-96 peak burnup is less than 25 MWd/kgU and the projected end-of-cycle (Cycle14) burnup is less than 28 MWd/kgU.

The impact of this change on peak cladding temperature is a small decrease. Specifically, the maximum peak cladding temperature changed 8° Fahrenheit (from 1909°F to 1901°F).

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1 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

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

The peak cladding temperature still remains well below the 2200° Fahrenheit acceptance limit.

In addition for the ABB fuel, the WNP-2 hydrogen generation calculation was revised to allow more direct verification for current and future operating cycles in both single and two loop operation. The revised analysis increased the reported results, which still remain well below the LOCA acceptance criterion.

Specifically, the reported hydrogen generation increased from less than 0.1 percent to less than 0.3 percent. This value is below the 1.0 percent limit specified in 10 CFR 50, Appendix K.

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

This section contains summaries of the Safety Evaluations (SE) completed for activities implemented during 1998 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 NRC approval, unless the proposed change, test or experiment involves a change in the Technical Specifications incorporated in the license or an unreviewed safety question.

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 which were within the total scope of the proposed change. 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

This section contains information pertaining to implemented Plant Modification Records (PMRs), Technical Evaluation Requests (TERs), and Temporary Modification Requests (TMRs) and is included pursuant to 10 CFR 50.59.

PMR 91-0071 (SE 98-043-00 and SE 98-043-01)

This modification provided for the replacement of residual heat removal system vent valves RHR-V-606 and RHR-V-631 with piping and associated connections at an existing three-quarter-inch sockolet weld on the low pressure coolant injection line to reduce susceptibility to fatigue induced failures.

Safety Evaluation Summary

It was concluded from the safety evaluation that the design function of the residual heat removal low pressure injection line remains unchanged as a result of this modification. This modification consists of a configuration that includes a pipe nipple, coupling and pipe plug that will increase piping connection strength and reduce the potential for fatigue failure.

With the removal of the two valves and modification of the branch connection, the reliability and integrity of the residual heat removal low pressure coolant injection function is enhanced and the potential for a small break LOCA is minimized.

The probability of occurrence of a previously evaluated accident or transient is not increased since the affected piping system and components were analyzed and qualified to the applicable ASME Code and WNP-2 design basis requirements. The safe shutdown analysis for pipe breaks inside of containment remains valid and is not impacted by this change.

PMR 91-0137 (SE 97-062)

This modification provided for the installation of a reactor closed cooling water system filter demineralizer and associated coupon stations to reduce the potential for fouling of safety-related valves.

Safety Evaluation Summary

It was concluded from the safety evaluation that the safety-related and non-safety related functions performed by the reactor closed cooling water system are unchanged by the modification. The only safety-related equipment associated with the reactor closed cooling water system are the containment isolation valves and those valves that interface with the standby service water system and the fuel pool cooling system heat exchangers..

The outlet of the demineralizer and the coupon stations were equipped with a resin trap to capture any potential material or loose parts. The reactor closed cooling water system is not associated with the reactor coolant pressure boundary and is not the initiator of any accident or transient.

The change does not have any affect on the limit for allowable temperature in the drywell or impact primary containment isolation valve operability. The change will not adversely impact accident and mitigation functions performed by the reactor closed cooling water system.

PMR 92-0120 (SE 93-199)

This modification provided for the replacement of the power fuses for standby liquid control system motor operated valve SLC-V-1A as part of a fuse coordination enhancement effort.

Safety Evaluation Summary

It was concluded from the safety evaluation that replacement of the power fuses ensures that the fuses are in conformance with sizing methods for safety-related valves. In addition, reducing the fuse size enhances the protection of the motor against a locked-rotor or locked-armature current condition.

Downsizing the fuses to 125 percent full load amps allows the motor operated valve to provide two full strokes under maximum degraded voltage, temperature and differential conditions without tripping. This change has no adverse impact on previously evaluated accidents or transients. Downsizing of the fuses provides increased motor protection and ensures that the valve is capable of performing its intended safety function.

PMR 92-0276-5F (SE 97-100-00, SE 97-100-01 and SE 97-100-02)

This modification provided new power feeders and related components such as cables, conduit, a local fused disconnect switch, conduit supports, fire-rated wrap for supports, and penetration sealant for residual heat removal system high/low pressure interface valves RHR-V-9, RHR-V-123A and RHR-V-123B.

The maintained-open disconnect switch for RHR-V-9 was also relocated from outside the reactor building to a compatible fire area in order for the switch to remain operable during accident conditions. Control circuit isolation from the main control room for RHR-V-8 was no longer required, which eliminated the need for a key-locked switch at the remote shutdown panel. Accordingly, the transfer switch is placed in the "normal" position during normal plant operation. The changes associated with these fire safe shutdown components were made as part of ongoing Thermo-Lag reduction efforts.

Safety Evaluation Summary

It was concluded from the safety evaluation that the re-routing of isolated power feeders in grounded conduit and protected supports, and the associated control and annunciator circuit changes for RHR-V-8 and RHR-V-9, does not result in any unanalyzed condition. The changes are bounded by existing analyses.

The system and component functions remain unchanged and are not impacted by the modification. The components involved were purchased for safety-related application and installed to Quality Class 1 or augment quality requirements in accordance with the component functions.

The Technical Specification requirements for isolation valves, high/low pressure interface valves, primary containment isolation capability, and motor-operated valve operability considerations are not affected by this change.

PMR 92-0276-07 (SE 97-122)

This modification provided for the elimination of post fire safe shutdown protection for residual heat removal system passive valves RHR-V-3B, RHR-V-4B, RHR-V-24B, RHR-V-27B and RHR-V-68B by providing alternative design features. The modification also provided for the elimination of "hot short" protection for motor operated spurious impact valves RHR-V-6B, RHR-V-16B, RHR-V-23, RHR-V-49, RHR-V-53B, RHR-V-115 and RHR-V-116 by reconnection of the control circuits. The changes associated with these fire safe shutdown components were made as part of ongoing Thermo-Lag reduction efforts.

Safety Evaluation Summary

It was concluded from the safety evaluation that post fire safe shutdown functions were preserved by the modification and the changes are bounded by existing analyses. The system and component functions remain unchanged and are not impacted by the modification.

The modification eliminates the need to credit post fire safe shutdown Thermo-Lag protection for several passive and spurious impact valves in the residual heat removal system. The passive valves are not required to change state for a fire event and the spurious impact valves are required to not rupture for a fire event. Since Thermo-lag combustibility has been incorporated into the combustibility calculations, the existing Thermo-Lag will remain in place.

Circuit protection against "hot shorts" is ensured by the re-routing of control cables into new Quality Class 1 raceways consisting of armored steel cable and conduit. The new raceways provide protection from fire-induced "hot short" cable failures by a ground path for any

external potential cross-connection of a power source to the control cables routed inside. This modification does not affect any other components and ensures that the control and operation of the affected residual heat removal system valves are unchanged.

PMR 92-0287 (SE 98-096)

This modification provided for the removal and replacement of the two non-functional air valves, on the non-automatic depressurization system main steam relief valves, with a single ported block.

Safety Evaluation Summary

It was concluded from the safety evaluation that removal of the two non-functional air valves and replacement with ported blocks does not affect the safety function of the main stream relief valves. The modification has no impact on any previously evaluated accident or transient.

The modification does not affect the function or operation of the non-automatic depressurization system relief valves. The ported block provides an equivalent air pathway as the replaced air valves. The probability and consequences of a malfunction of the affected relief valves remains unchanged by the modification. The modification reduces the number of potential leakage paths in the system, while maintaining the pressure integrity. The ported blocks have no moving parts that could be susceptible to malfunction or failure.

PMR 93-0046 (SE 96-072)

This modification provided for the installation of an upgraded computer system for the Plant Data Information System (PDIS). Specifically, the Emergency Response Data System (ERDS) was relocated to a new, functionally equivalent computer system connected to the control room local area network.

Safety Evaluation Summary

It was concluded from the safety evaluation that the software and hardware modifications would have no impact on any accident/transient or equipment malfunction analyses. The activity does not change the function of the PDIS or the ERDS.

The modification simply relocated the ERDS function from a Prime computer system to a functionally equivalent and Y2K compliant computer system connected to a control room local area computer network.

Power to the new equipment continues to be supported by a battery-backed uninteruptible power supply and an emergency diesel generator. Calculations have shown that the added loads do not impact the availability of power for safety-related systems or the ability of the control room cooling system to maintain temperatures within design basis limits.

PMR 94-0043 (SE 97-097-00 and SE 97-097-01)

This design change provided for modification of the sealing mechanisms for residual heat removal system valve RHR-V-42A and high pressure core spray system valve HPCS-V-15 to provide for venting of pressure between the wedge discs to the high pressure side (containment side) of the valve. For RHR-V-42A, an external bypass line was added which equalizes the pressure between the wedge disc and the reactor side of the valve ends. For HPCS-V-15, a bypass line was installed downstream of HPCS-V-85. A new vent valve was also added to replace the isolation function of HPCS-V-85.

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.

This activity was a commitment made in response to NRC 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 coolant injection.

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

PMR 94-0084 (SE 98-011)

This modification provided for the re-routing of piping and the installation of pumps to support the addition of sodium bromide for the treatment of both the circulating water system and plant service water system to improve water quality treatment and prevent further corrosion in system piping.

Safety Evaluation Summary

It was concluded from the safety evaluation that the addition of sodium bromide in the circulating water system and plant service water system would have no impact on any accident/transient or equipment malfunction analyses. The circulating water system and the

plant service water system are not initiators or mitigators of design basis accidents or malfunctions.

The change to the general addition of chemicals as described in design basis documentation does not alter current analyses. This modification does not interface with any safety-related structure, system or component.

The margin of safety as defined in the basis for any Technical Specification is not decreased by the improved biological fouling control in the circulating water system and the plant service water system. The only adverse impact could be the potential for exposure to short periods of degraded water quality resulting from a main condenser tube leak. However, water quality is continuously monitored and limits and associated corrective actions have been established in the event that water quality parameters become degraded.

PMR 94-0128 (SE 97-088)

This modification provided for the re-routing of a drain-to-condenser line associated with reactor feedwater system Heater 6A. The line was routed to a location below the centerline of Heater 6A to reduce the potential for flow transients such as water hammer.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification has no impact on the function and operation of the reactor feedwater heater drain system. The heater drain piping system and all components, including piping supports, are designed and qualified to meet applicable codes and standards.

The modification has no impact on any previously evaluated accident or transient. The failure of pipe supports was considered and determined not to be credible events because they were designed to withstand normal and faulted loads. A pressure boundary failure was also determined to be a non-credible event because stress analysis of the piping system showed adequate margin and conformance to the Power Piping Code.

There is no change in the function and operation of reactor feedwater system Heater 6A. The startup procedure for Heater 6A will be simplified by the change. There is no change in the configuration or function of any equipment.

PMR 94-0274 (SE 97-105)

This modification provided for the installation of a small bore piping tap, with isolation valves, in the suction and discharge of the condensate booster pumps. In addition the modification provided for a small bore piping tap, with isolation valves, in each train of the

off-gas system upstream of the recombiners. The taps were installed in preparation for the planned implementation of a hydrogen water chemistry system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the condensate and off-gas systems serve no function in mitigating the consequences of a postulated design basis accident or anticipated operational transient. The condensate and off-gas systems are not safety-related, do not perform any safety-related functions, and do not affect any safety-related functions. The new piping taps are no different than the vent and drain connections that are currently installed in the condensate system. The new valves are similar to those already installed in the systems.

Piping connections were designed in accordance with the classifications for the piping systems (i.e., Quality Class 2/Seismic Class II for the condensate system and Quality Class 2+/Seismic Class II for the off-gas system).

The piping taps in the off-gas system will not affect the margin of safety in the Technical Specifications for maintaining gross gamma activity of noble gases at the condenser air ejectors below a preset limit. The piping is designed to contain an internal detonation in the off-gas system.

PMR 95-0023 (SE 97-142)

This modification provided for the installation of a new sparger assembly associated with radioactive floor drain system waste sludge separator tank FDR-TK-22 to improve the circulation of spent resin.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification does not affect the function of FDR-TK-22. The tank is not credited with being an accident initiator and the change does not add an interface to a safety-related system or change the operation of any systems or components.

The mixing capability of the sparger system will be enhanced by this design change. The modification of the sparger assembly inside the tank does not change the operation of the tank or any of its components, nor does it introduce a unique failure mechanism into the system.

PMR 95-0023-01 (SE 98-010)

This modification provided for the installation of a Y-type strainer between radioactive floor drain pump FDR-P-23 and the sparger assembly associated with radioactive floor drain system waste sludge separator tank FDR-TK-22 to prevent sparger failure due to clogging of the nozzles.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification does not affect the function of the radioactive floor drain processing system. The system is not credited with being an accident initiator and the addition of the strainer does not add an interface to a safety-related system or change the operation of any systems or components.

The function of the sparger system will be enhanced by this design change. The modification does not change the operation of the tank or any of its components, nor does it introduce a unique failure mechanism into the system. The installation of the strainer will have no adverse impact on the safe operation of the plant. From an ALARA perspective, radiation exposure will be reduced by decreasing the maintenance activities associated with this part of the system.

PMR 95-0236 (SE 97-103)

This modification provided for the replacement of nine top-entry local power range monitor detector assemblies with bottom-entry models of an improved design. This modification was made to provide for improved efficiencies during refueling outages.

Safety Evaluation Summary

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

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

PMR 95-0242 (SE 96-084)

This modification provided for the addition of longer filter septa in the condensate demineralizer tanks to reduce flow rate and, therefore, increase resin performance.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change has no impact on water chemistry limits and will not increase allowable condensate demineralizer effluent impurity levels or increase allowable contribution to activity levels. The modification also does not decrease condensate filter demineralizer ion exchange efficiency or change the means by which condensate filter demineralizer performance is monitored.

By maintaining the same administrative and regulatory standards for water chemistry and maintaining the currently prescribed routine water sampling and analysis, there is no increase in the exposure of any safety-related equipment to adverse water chemistry as a result of normal or faulted operation of the condensate filter demineralizers.

PMR 96-0139 (SE 97-087)

This modification provided for the installation of ten large capacity passive emergency core cooling system pump suction strainers in the suppression pool. The strainers were installed in response to concerns outlined in NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors." Specifically, industry problems were identified with maintaining adequate suction head for emergency core cooling system pumps due to excessive pressure drops across plugged strainer beds.

Safety Evaluation Summary

It was concluded from the safety evaluation that the function of the emergency core cooling system pump suction strainers remains unchanged. The replacement strainers were designed using a methodology endorsed by the NRC and the Boiling Water Reactor Owners' Group. The strainer design for head loss was substantiated by analysis and prototype tests which modeled post-accident debris load in the suppression pool. The debris loading for the WNP-2 suppression pool has been established by analysis and was used for the design basis for the strainers. The debris loading analysis was also used for the determination of available net positive suction head for emergency core cooling system pump performance under design basis accident conditions.

The new suction strainers were designed to prevent the introduction of solid particles greater than 3/32-inch into the emergency core cooling system flow stream. This function protects the emergency core cooling system pumps from damage and prevents the clogging of spray nozzles in the reactor, drywell and suppression pool.

The strainers were designed to perform satisfactorily in the presence of 100 percent of the containment coatings which are installed in the loss-of-coolant accident pipe break steam/water jet zone of influence. The replacement strainers will not impact design flow rates and pressures in the emergency core cooling systems. Air entrainment (i.e., vortexing) is not a concern with the new design because no changes were made to flow characteristics, pipe sizes or submergence levels.

PMR 96-0140 (SE 98-009)

This modification provided for the removal of test valves RRC-V-915 and RRC-V-916 in the reactor recirculation system and the installation of flanges to support replacement of valve RRC-V-20.

Safety Evaluation Summary

It was concluded from the safety evaluation that removal of the test valves has no impact on reactor recirculation system operation. The test valves performed no active system function and were added as a temporary solution for the purposes of evaluating leakage from a valve. The valves were normally closed and capped.

The removal of the valves will restore system piping to its original configuration. The function and operation of the reactor recirculation system is not changed by this modification.

PMR 96-0210 (SE 98-064)

This modification provided for the installation of a leak rate test globe valve and spectacle flange on a two-inch demineralizer water system supply line upstream of primary containment isolation valve DW-V-156. The modification will allow for the positive mechanical isolation of demineralized water flow at and through primary containment penetration X-92.

Safety Evaluation Summary

It was concluded from the safety evaluation that this modification does not impact primary containment functional integrity during and following peak transient pressure and temperatures that would be expected from any postulated design basis accident.

The modification provides a permanent resolution to a concern expressed in NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions." The concern was that, during post-LOCA conditions, containment penetrations with inboard and outboard isolation valves are susceptible to failure by thermal overpressurization when leak tight and filled with water.

Normally, the subject piping will remain drained and filled with air and not water at all times, except when demineralized water supply to primary containment is needed for maintenance activities during Mode 4 or 5.

PMR 97-0017 (SE 97-095)

This modification provided for the installation of a pressure bypass line around reactor closed cooling water system valve RCC-V-40 to limit pressure in the section of piping between containment isolation valves RCC-V-21 and RCC-V-40. A spring-loaded check valve (RCC-V-219) and two test valves (RCC-V-220 and RCC-V-221) were included in the bypass line boundary.

Safety Evaluation Summary

It was concluded from the safety evaluation that installation of the bypass line will enhance the safety of the reactor closed cooling water system return line containment penetration by limiting pressure in the line. Check valve RCC-V-219 will function as an inboard containment isolation valve in parallel with RCC-V-40. The valve was oriented against the normal flow in the reactor closed cooling water system and its purpose is to open momentarily during a loss of coolant accident to vent fluid between RCC-V-21 and RCC-V-40.

The addition of the bypass line does not impact the function or operation of the reactor closed cooling water system. The bypass line improves pressure boundary protection capability by ensuring that the pressure in the containment piping does not exceed the hydro-test pressure of the pipe. The impact of the potential for system leakage through the containment penetration will be established by the local leakrate testing process. The inboard isolation capability will be demonstrated by the inservice testing program.

PMR 97-0030 (SE 97-091)

This modification provided for the relocation of the low pressure alarm function from an existing residual heat removal system pressure indicating switch (RHR-PIS-22C) to a new device (RHR-PS-22C). The existing pressure switch provided both the high and low pressure alarm functions. The change was made to enhance the reliability of the alarm function.

Safety Evaluation Summary

It was concluded from the safety evaluation that installation of new residual heat removal system pressure switch RHR-PS-22C will not impact the performance or safety function of any plant system. The switch is Quality Class 1 since it performs a passive safety-related mechanical function. The low alarm range capability of the new component will allow for reliable operation with minimal drift.

The new pressure switch provides the same low pressure alarm function. The new component has a more compatible range to the low pressure setpoint. Therefore, a reduction in calibration frequency and associated personnel radiation exposure is expected.

The switch is fully qualified for its safety-related application and the modification will have no impact on the plant electrical distribution system because the same power source will be used. The modification will have no physical interface with any other system.

PMR 97-0040-01 (SE 97-093-00)

This design change provided for implementation of the WNP-2, Cycle 14 core design. The proposed activity consisted of operation of the Cycle 14 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).

Safety Evaluation Summary

It was concluded from the safety evaluation that operation of Cycle 14 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.

PMR 97-0040-01 (SE 97-093-01)

This revised design change provided for allowing the use of normal scram speed minimum critical power ratio limits with up to eight "slow" control rods.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that allowance for up to eight slow control rods in both Normal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) is accounted for in the transient analysis performed by the fuel vendor using NRC-approved methodology.

Operation with up to eight slow control rods in either NSS or TSSS is allowed by the Technical Specifications and will not affect core thermal limits and, therefore, will not impact the margin of safety provided their impact is analyzed with approved methodology. The technical basis and analytical assumptions for allowing slow control rods when using NSS limits is the same as when using the TSSS limits.

PMR 97-0059 (SE 97-138)

This modification provided for the installation of two normally-open isolation gate valves (TSW-V-302 and TSW-V-303) and two normally-closed drain ball valves (TSW-V-782 and TSW-V-783) to allow for isolation and drainage capability for the entire radwaste building plant service water system load.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The plant service water system is a non-safety related system and is not required to perform any safety function or ensure: 1) the integrity of the reactor coolant boundary; 2) the capability to shutdown the reactor and maintain it in a safe shutdown condition; and 3) the ability to prevent or mitigate the consequences of accidents.

The piping design continues to meet all system design requirements and system function remains unchanged by the modification. Therefore, there is no adverse impact on associated piping stress and support evaluations. Installation of the modification also does not impact existing plant service water pipe break analyses.

Installation of the two isolation and drain valves increases the reliability of the plant service water system by allowing for maintenance activities in the radwaste building that would otherwise require a full system outage.

PMR 97-0060 (SE 97-139)

This modification provided for the installation of two normally-open isolation gate valves (TSW-V-304 and TSW-V-305) and two normally-closed drain ball valves (TSW-V-784 and TSW-V-785) to allow for isolation and drainage capability for the plant service water system load for main turbine oil heat exchangers TO-HX-1A and TO-HX-1B.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The plant service water system is a non-safety related system and is not required to perform any safety function or ensure: 1) the integrity of the reactor coolant boundary; 2) the capability to shutdown the reactor and maintain it in a safe shutdown condition; and 3) the ability to prevent or mitigate the consequences of accidents.

The piping design continues to meet all system design requirements and system function remains unchanged by the modification. Therefore, there is no adverse impact on associated piping stress and support evaluations. Installation of the modification also does not impact existing plant service water pipe break analyses.

Installation of the two isolation and drain valves increases the reliability of the plant service water system by allowing for maintenance activities in the radwaste building that would otherwise require a full system outage.

PMR 97-0076 (SE 98-053)

This modification provided for the installation of a blind flange on the downstream face of circulating water system blowdown valve CBD-V-3 to prevent leakage and, therefore, potentially uncontrolled liquid effluent discharge to the Columbia River. Valve CBD-V-3 is a butterfly valve that is installed on a 24-inch bypass line to CBD-LCV-1. (Flow control valve CBD-CFV-1 is the flow level controller for the liquid effluent discharge line to the river.)

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification has no affect on any previously evaluated accident or transient. The circulating water blowdown system is not required for safe shutdown and has no safety or safety support functions. The existing equipment and piping system meet applicable codes and WNP-2 design specifications.

Valve CBD-V-3 is a bypass around normal plant liquid effluent discharge valve CBD-LCV-1. The circulating water blowdown system piping system is of a low energy, low pressure and low temperature design.

The consequences related to a circulating water blowdown system component malfunction are associated with the uncontrolled release of liquid radioactive waste. However, protection against such an event continues to be provided by existing design redundancy, instrumentation for detection, alarm of abnormal conditions, and procedural controls.

PMR 97-0103 (SE 97-079)

This modification provided for the replacement of the motor for the Parson Peebles circulating water system pump CW-P-1C with a new Westinghouse spare motor. This change also consisted of modification of the associated cooling lines and electrical terminations. The winding insulation on the original motor had failed. (The physical work was completed in 1997, with final modification close-out occurring in 1998.)

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on the operation of the circulating water system or any other plant system. The circulating water system is not required to maintain the reactor in a safe shutdown condition or mitigate the consequence of an accident.

The change was simply limited to the replacement of the existing motor with a new spare motor. Both the original motor and the replacement motor have identical mechanical and electrical performance ratings. The function of circulating water system pump CW-P-1C is unchanged by the modification. The modification does not impact any transient analysis or affect the operation of any safety-related or important-to-safety equipment.

PMR 97-0148 (SE 98-060)

This modification provided for the replacement of radioactive equipment drain system transmitter EDR-FT-37 with an ultrasonic flow transmitter, which is designed to monitor low flow conditions.

Safety Evaluation Summary

It was concluded from the safety evaluation that the radioactive equipment drain system is not required for design basis accident safety functions, nor does it have any direct interface with safety-related systems. The modification does not impact any accident or transient analysis or affect the operation of any safety-related or important-to-safety equipment.

The design change does not impact the function or operation of the radioactive equipment drain flow monitoring system. The replacement transmitter is an ultrasonic flow monitor which is designed to monitor this type of application. The flow monitor is unintrusive and is strapped to the outside of the pipe. Replacement of the existing flow transmitter with another type of component that was specifically designed for this application will enhance flow measurement.

Installation requirements were based on electrical separation and seismic design criteria. Combustible load increases due to the modification were also evaluated and found to be acceptable. The new flow transmitter will maintain the range of radioactive drain flow monitoring required by the Technical Specifications.

PMR 97-0186 (SE 98-083)

This modification provided for the replacement of centrifugal bulk chemical feed system pump CF-P-43 with a positive displacement metering-type pump that is capable of injecting chemicals at a continuous lower flow rate into the circulating water system and plant service water system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The circulating water system and the plant service water system are non-safety related systems and are not required to perform any safety function or ensure: 1) the integrity of the reactor coolant boundary; 2) the capability to shutdown the reactor and maintain it in a safe shutdown condition; and 3) the ability to prevent or mitigate the consequences of accidents.

The change to the chemical feed system as described in design basis documentation does not alter current analyses. This modification does not interface with any safety-related structure, system or component.

Pump CF-P-43 is primarily used to feed a copper corrosion inhibitor into the circulating water system and the plant service water system. This modification will provide for improved copper corrosion control and also assist in ensuring compliance with copper discharge limits prescribed by the National Pollution Discharge Elimination System permit for WNP-2.

PMR 97-0274 (SE 98-088)

This modification provided for the replacement of the existing hydrogen analyzer panels with two new hydrogen/oxygen analyzers in preparation for the planned implementation of a hydrogen water chemistry system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modification has no affect on any previously evaluated accident or transient. The off-gas hydrogen analyzer system is non-safety related. The new analyzers will perform the same basic function as the existing units.

The explosive gas monitoring system is not used for, nor is capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary. The explosive gas monitoring system is not used to monitor any process variable that is an initial condition of a design basis accident. Excessive system hydrogen is not an indication of a design basis accident or transient and the explosive gas monitoring system is not part of a primary success path in the mitigation of an accident or transient.

The potential for system faults due to moisture sensitivity will be eliminated since the new sensors are unaffected by moisture. The time required to calibrate the new analyzers is much shorter and the time span between calibrations is greatly increased, therefore, out-of-service time will be reduced.

TER 96-0214 (SE 97-009)

This modification provided for the installation of a check valve in the fill line to circulating water system sulfuric acid tank SAT-TK-2 to prevent the possibility of draining the tank if the transfer hose becomes disconnected from the fill station.

Safety Evaluation Summary

It was concluded from the safety evaluation that the addition of a check valve in the acid fill station does not impact the integrity of the reactor coolant boundary or the capability to shutdown the reactor and maintain it in a safe shutdown condition. The sulfuric acid system is a non-safety related system which does not perform any safety function.

The addition of a check valve in the fill connection improves personnel safety considerations for individuals involved with the transfer of acid. The addition of the check valve also reduces the impact of environmental considerations in the unlikely event the fill hose becomes disconnected during filling of the sulfuric acid tank.

TER 97-0104 (SE 97-085)

This modification provided for the replacement of reactor recirculation system gate valves RRC-V-8A and RRC-V-8B with globe valves to provide for improved flow control when restoring seal purge at high differential pressures. The valves serve as a manual shutoff to the seal surge supply piping for reactor recirculation system pumps RRC-P-1A and RRC-P-1B.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on the flow or pressure rating of the affected piping system. The replacement valves meet design requirements for material and pressure rating and the associated performance characteristics of the affected lines will not be degraded.

The replacement activity consisted of an equivalent change that only impacted non-ASME, non-safety related Quality Class 2 components. The weight of the replacement globe valves is significantly less than the existing gate valves, therefore, piping and piping support stress levels were reduced by this modification.

TER 97-0118 (SE 97-119)

This modification provided for the replacement of plant service water system valves TSW-V-272 and TSW-V-273 with threaded pipe caps to decrease the potential for galvanic corrosion of nearby carbon steel piping in the system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The plant service water system is a non-safety related system and is not required to perform any safety function or ensure: 1) the integrity of the reactor coolant boundary; 2) the capability to shutdown the reactor and maintain it in a safe shutdown condition; and 3) the ability to prevent or mitigate the consequences of accidents.

Removal of brass-bodied valves TSW-V-272 and TSW-V-273 will decrease the probability of through-wall piping corrosion in the plant service water system and the recurrence of galvanic corrosion in the area where the valves were located. The weight of the threaded pipe caps is significantly less than the weight of the valves. Therefore, there is no adverse impact on associated piping stress and support evaluations. Installation of the modification also did not impact existing plant service water system pipe break analyses.

TER 98-0006 (SE 98-051)

This modification provided for the replacement of standby service water system combination switch SW-FS-19 and SW-FI-19 with a single differential pressure indicating switch (SW-FIS-19) to improve component reliability. The deadband of SW-FS-19 was such that, at normal flow rates, the device did not reliably reset. The SW-FS-19 and SW-FI-19 combination functioned as the low flow alarm signal to the main control room for the low pressure core spray system pump motor upper bearing cooler.

Safety Evaluation Summary

It was concluded from the safety evaluation that the alarm for the low pressure core spray system pump motor upper bearing cooler is not an initiator of any previously evaluated accidents or transients.

The safety function for the flow indicating device is passive pressure integrity. The new component has a higher pressure rating than the existing devices (2500 psig versus 1500 psig). Therefore, the new single differential pressure indicating switch will not adversely affect the component safety function.

Replacing two safety-related devices with one component reduces that probability of equipment malfunction. Based on plant operating experience, the differential pressure indicating switch has a predicted high degree of reliability. The component will also provide a more accurate measure of standby service water system flow through the low pressure core spray system pump motor upper bearing cooler during the performance of operational surveillances.

TER 98-0055/TER 98-0056 (SE 98-075)

This modification provided for the addition of hard piping vent lines from the top of residual heat removal system heat exchangers RHR-HX-1A and RHR-HX-1B to allow plant operators to vent the heat exchanger piping without having to enter a high radiation/contamination zone.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The new piping has been designed and analyzed in accordance with ASME III, Class 2, piping requirements. The new pipe supports were designed and analyzed in accordance with Seismic Category I criteria.

All of the new components are ASME Class III material which meets code specifications for the piping and supports. Accordingly, there is no impact on the safety function of the vent lines. The new piping has been designed to maintain the pressure integrity of the system.

TMR 98-020 (SE 98-072)

This modification provided for the installation of equipment to maintain a small nitrogen gas cushion on several fire protection system standpipes to reduce the impact velocity of water surges caused by system dynamics.

Safety Evaluation Summary

It was concluded from the safety evaluation that maintaining a small cushion of nitrogen gas at the top of the standpipe riser has no impact on any previously analyzed accident or transient.

The modification has no impact on the ability of a standpipe to perform its design function. The standpipe and fire protection hose system will continue to be available if needed for manual fire fighting. The addition of the nitrogen cushions is not in conflict with any fire protection design basis requirement. The modification does not increase the probability of a failure of the water supply associated with the fire protection system.

TMR 98-021 (SE 98-073)

This modification allowed for the temporary closure of radioactive floor drain system valves FDR-V-607, FDR-V-608 and FDR-V-609. These valves isolate the sump in one emergency core cooling system pump room from the drain lines in another pump room. (After the fire protection valve break and subsequent flooding of residual heat removal system RHR-P-2C pump room in June 1998, FDR-V-609 did not close on a high-high sump level signal, nor did it fail closed as designed.)

Safety Evaluation Summary

It was concluded from the safety evaluation that closure of valves FDR-V-607, FDR-V-608 and FDR-V-609 places them in the position expected in design basis flooding calculations. Closure of the valves does not create, increase the probability of, or increase the consequences of any previously evaluated transient or accident.

Normal drainage of the low pressure core spray system pump room, the reactor core isolation cooling system pump room and the control rod drive/condensate system pump room will require periodic operator attention to empty the isolated drain lines of any accumulated water. In addition, there is the potential of small leaks occurring that cause water to back up into the drains faster than expected. Therefore, the probability of water accumulating on the floor in those rooms is increased when these valves are closed.

However, because of other equipment available to compensate for the functions provided by the radioactive floor drain system, the probability of malfunction of equipment important to safety is not increased.

2.6.2 Licensing Document Changes

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

During 1998 the WNP-2 FSAR Upgrade Project was completed. The FSAR Upgrade Project LDCNs are identified as such below. In general, the FSAR Upgrade Project LDCNs include many enhancements, clarifications, and corrections to the WNP-2 FSAR. The most significant enhancements include the following:

- The FSAR was reformatted to provide uniformity in format, nomenclature, grammar and punctuation.

- Drawings in the FSAR were consolidated to eliminate multiple figures that depicted the same system drawing.

- Appendix D, "Questions and Responses," was eliminated. The original questions and responses that were submitted remain in the docket file. The responses to the questions were reviewed and, if not already included, were appropriately incorporated into the body of the FSAR.

- Some information was annotated as "historical" using the guidance in NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 0.

The following is a discussion of those FSAR Upgrade Project and routine LDCNs that were approved for implementation during 1998.

LDCN-FSAR-97-060 (SE 97-120)

FSAR Upgrade Project LDCN, Section 10.3, "Main Steam Supply System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the main steam system drain lines.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that deletion of the discussion pertaining to the drain line low point location and routing to the main condenser is basic design information that does not pertain to any safety function or related structure, system or component. In addition, the safety-related containment isolation function for the main steam drain lines is adequately discussed in FSAR Chapter 6, "Engineered Safety Features," and radiological release points and values are maintained in Chapter 11, "Radioactive Waste Management."

LDCN-FSAR-97-061 (SE 97-106)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the main condenser sampling system.

Safety Evaluation Summary

It was concluded from the safety evaluation that revising the FSAR to reflect that the original condenser conductivity sampling system is not used, and that a conductivity/chemical species sampling and measurement system is available for characterizing condenser in-leakage, is an overall improvement from the original equipment currently described in the FSAR.

This change affects only the method used to characterize conductivity for condenser in-leakage monitoring and identification and improves the capability to detect and correct any condenser in-leakage. This change does not otherwise impact any structure, system or component.

LDCN-FSAR-97-061 (SE 97-108)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with low pressure turbine gland seal pressure and reactor feedwater pump turbine exhaust seal pressure.

Safety Evaluation Summary

It was concluded from the safety evaluation that changing the low pressure turbine gland seal description by replacing the value of 16 psia seal steam supply pressure with the term "sufficient sealing steam to maintain positive seal pressure" would not alter the function of the steam seals. A similar change was also made to the reactor feedwater pump turbine exhaust seal pressure (from "18 to 21 psia" to "sufficient to maintain positive seal pressure"). The function of the steam seals is to prevent in-leakage. Since the main turbine low pressure and feedwater pump turbine exhaust pressures are at a vacuum, this function is accomplished as long as a positive pressure is maintained.

This change does not alter the function or operation of these gland seals. Removal of the currently-specified value prevents either excessive system calibration or operability questions if the supply pressure is not exactly 16 psia for the low pressure turbine and 18 to 21 psia for the feedwater pump turbine.

LDCN-FSAR-97-061 (SE 97-109)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the turbine gland sealing system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting the description of the gland steam condenser tube plugging capability during operation is not related to the design function of the system or any other system component.

The capability to perform tube repair on-line is related to continued power generation only and is not associated with any accident or malfunction response or mitigation capability. Operation of one blower during plant operation is retained in FSAR Section 10.4.3.2, "Turbine Gland Sealing System Description."

LDCN-FSAR-97-061 (SE 97-113)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with condensate filter demineralizer system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting the description of the specific material type (nylon wound) for the filter demineralizer precoat filter elements does not affect the design function of the system or the filter demineralizers as long as the necessary design characteristics to support the capability to maintain water chemistry within limits are maintained.

Any future changes to the filter element material type would continue to require consideration of potential impact on the described design functions of the filter demineralizer system.

LDCN-FSAR-97-061 (SE 97-116)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the main turbine gland sealing system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting a description related to providing clean steam to the main turbine gland sealing system prior to starting the vacuum pumps does not result in an unreviewed safety question. The only relevance of supplying main turbine gland sealing steam prior to starting the mechanical vacuum pumps pertains to potential adverse effects of drawing cooler air through the gland and over a hot turbine shaft if the turbine is above ambient temperature and sealing steam is not provided.

The only potential adverse effects of drawing cool air over a hot shaft would be possible shaft surface cracking. Surface cracking is a slow initiation and propagation phenomenon that would be detected during periodic inspections and corrected prior to developing to a point that would create a concern for potential adverse effects. Current procedures control starting the vacuum pumps without gland sealing steam as long as the turbine is not above ambient temperature.

LDCN-FSAR-97-061 (SE 97-117)

FSAR Upgrade Project LDCN, Section 10.4, "Other Features of Steam and Power Conversion Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the main condenser.

Safety Evaluation Summary

It was concluded from the safety evaluation that any variations to the original design values for the main condenser are inconsequential provided the functional requirements for the main condenser are maintained. This change added "approximate values" to the condenser design conditions, deleted circulating water velocity, and revised total steam condensed and total condensate outflow values to reflect a previous power uprate licensing change. In addition, the duty value was rounded off consistent with the change to reflect "approximate values."

The approximate values are retained in the FSAR as a general description of the main condenser. The velocity in the tubes is a level of detail beyond that which is useful and is also inconsequential as long as the functional requirements for the main condenser are maintained. The total steam condensed and total condensed outflow value changes were

previously evaluated as part of the previous power uprate licensing change. Rounding off the design value for the main condenser is consistent with the change to reflect that the values are approximate.

LDCN-FSAR-97-109 (SE 98-076)

FSAR Upgrade Project LDCN, Section 2.3, "Meteorology." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the meteorological system.

Safety Evaluation Summary

It was concluded from the safety evaluation that revising the FSAR to eliminate information about WNP-1, WNP-4, and the Fast Flux Test Facility (FFTF) is appropriate since WNP-1 and WNP-4 are terminated projects and FFTF installed its own meteorological tower, so information is no longer required from the WNP-2 meteorological system.

The remaining information satisfies all requirements for a meteorological system. The differential temperature measurement instrumentation has shown that it is reliable and performs within the accuracy requirements for the system. The computer floppy disks perform the same function as the magnetic tape system. The storage of the data on the floppy disks is not a requirement since plant computers also store the same information.

LDCN-FSAR-97-117 (SE 97-133)

FSAR Upgrade Project LDCN, Section 3.5, "Missile Protection." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with standby service water system and tower makeup system tornado protection.

Safety Evaluation Summary

It was concluded from the safety evaluation that revising the standby service water system and tower makeup system underground piping and electrical line tornado protection description to replace "sufficient Quality Class 1 earth cover of high relative density" with "sufficient earth cover" for the tower makeup system does not alter the standby service water system earth cover description. Tower makeup system backfill was neither originally, nor is it currently, Quality Class 1.

Tower makeup system supply to the spray ponds is credited to refill the ponds in the event spray pond water is lost due to a tornado and to supply cooling water for feed and bleed cooling in the event spray headers are damaged by a tornado missile. A postulated tornado

is the only design basis event, accident, transient, or malfunction that the tower makeup system-to-spray pond makeup function is credited for mitigation.

Since the tower makeup system is not safety-related, the underground utilities are not required to be Quality Class 1. The only required design function of the earth cover for underground tower makeup system piping and electrical lines, other than standard support and freeze protection, is to provide protection from tornado missiles. Based on analysis of the maximum calculated missile penetration depth and the design of the tower makeup system underground utilities, the utilities will not be damaged by the postulated design basis tornado missiles.

LDCN-FSAR-97-120 (SE 97-123)

FSAR Upgrade Project LDCN, Sections 5.4.3, "Reactor Coolant Piping," 5.4.4, "Main Steam Line Flow Restrictors," 5.4.5, "Main Steam Line Isolation," and 5.4.9, "Main Steam Lines and Feedwater Piping." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with main steam isolation valve testing.

Safety Evaluation Summary

It was concluded from the safety evaluation that revising the main steam isolation valve (MSIV) testing description does not alter the original intent since the described testing is more accurately referred to as leakage tests. Increasing the specific pressure at which the MSIV pressure boundary leakage test is performed does not alter the intended purpose of the test as long as the valve body, not valve seat, receives the same leakage test as is required for the primary system.

The primary system and MSIV pressure boundary leakage tests are performed in accordance with applicable code requirements which ensures pressure boundary integrity during startup following extensive shutdowns. Increasing the specified pressure for the MSIV pressure boundary leakage test does not alter the intended purpose of the testing or impact the function of the MSIVs or any other structure, system or component.

LDCN-FSAR-97-124 (SE 97-125)

FSAR Upgrade Project LDCN, Section 5.4.6, "Reactor Core Isolation Cooling System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the reactor core isolation cooling system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting steps used to describe specific operation of the reactor core isolation cooling system is an administrative change to remove extraneous information that is not necessary. As stated in the FSAR, the procedure steps being deleted "are not all the steps of the procedure, but give an example of most of the actions."

Any changes to the information being deleted could only impact the facility or processes, as described in the FSAR, if the change physically impacted the reactor core isolation cooling system or the capability of the reactor core isolation cooling system to perform its intended design function. Any such changes would require evaluation against the system design and functional requirement information that is being retained in the FSAR. This change has no physical impact on the function of the reactor core isolation cooling system.

LDCN-FSAR-97-127 (SE 97-131)

FSAR Upgrade Project LDCN, Section 5.4.7, "Residual Heat Removal System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the residual heat removal system.

Safety Evaluation Summary

It was concluded from the safety evaluation that replacing specific residual heat removal system operation details that minimize the potential for exceeding the 100° Fahrenheit per hour Technical Specification cooldown limit with "precautions and limitations in the appropriate operating procedures" does not alter the original intent of the FSAR to describe how the potential for exceeding the cooldown rate limit is minimized.

The existing 80° Fahrenheit per hour administrative cooldown rate limit is specified in the precautions and limitations of the normal shutdown and residual heat removal system operating procedures which supports the revised FSAR description.

LDCN-FSAR-97-127 (SE 98-047)

FSAR Upgrade Project LDCN, Section 5.4.7, "Residual Heat Removal System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the residual heat removal system.

Safety Evaluation Summary

It was concluded from the safety evaluation that operation of residual heat removal system "A" or "B" in secondary modes, specifically suppression pool cooling and wetwell sprays; or operation of residual heat removal system "C" in the full-flow test/mixing mode, does not render the residual heat removal low pressure coolant injection function inoperable.

When operated in these modes, the residual heat removal system will reliably respond to LOCA signals and be available to support the low pressure coolant injection function in accordance with the LOCA analysis assumptions. Similarly, the low pressure core spray and high pressure core spray systems, when they are operated in the suppression pool test return modes, will reliably respond to LOCA signals and be available to support the core spray functions in accordance with LOCA analysis assumptions.

Operation in secondary modes requires the positioning of some emergency core cooling system valves to non-accident positions, however, the valves will reliably reposition in accordance with the design basis to support respective system safety functions.

LDCN-FSAR-97-128 (SE 97-127)

FSAR Upgrade Project LDCN, Sections 5.4.8, "Reactor Water Cleanup System," 5.4.12, "Pressure Relief Discharge System Valves," 5.4.13, "Safety and Relief Valves," and 5.4.14, "Component and Piping Supports." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the reactor water cleanup system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the elimination of the description of the time required to perform a reactor water cleanup system filter demineralizer backwash/precoat operation does not impact the capability of the system to function as required as long as water chemistry can be maintained within specified limits.

Any changes that could impact the capability to maintain water chemistry as described in FSAR Sections 10.4.6, "Condensate Filter Demineralizer System," and 5.2.3, "Reactor Coolant Pressure Boundary Materials," would continue to require consideration of potential impact on the FSAR water chemistry description.

LDCN-FSAR-97-128 (SE 97-128)

FSAR Upgrade Project LDCN, Sections 5.4.8, "Reactor Water Cleanup System," 5.4.12, "Pressure Relief Discharge System Valves," 5.4.13, "Safety and Relief Valves," and 5.4.14, "Component and Piping Supports." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the reactor water cleanup system filter demineralizers.

Safety Evaluation Summary

It was concluded from the safety evaluation that the deletion of the specific material type of ion exchange resins from the filter demineralizer description cannot impact the design function of the reactor water cleanup system filter demineralizers as long as the necessary design characteristics to support the capability to maintain water chemistry within limits are maintained.

Any future changes to the ion exchange resin type would continue to require consideration of potential impact on the design functions of the reactor water cleanup system.

LDCN-FSAR-97-134 (SE 98-004)

FSAR Upgrade Project LDCN, Section 9.2.1, "Plant Service Water System," 9.2.2, "Reactor Building Closed Cooling Water System," 9.2.3, "Plant Makeup Water Treatment and Demineralized Water System," and 9.2.4, "Potable Water and Sanitary Drain Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with operation of the potable water and sanitary drain systems.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting the potable water and sanitary drain system description, testing, and instrumentation details from the FSAR does not affect any margin of safety. Spray pond makeup is not considered safety-related, except in the event of water loss due to a tornado. In the event of a tornado, the tower makeup system is the analyzed makeup source.

Other than spray pond makeup, the potable water and sanitary drain system does not service any safety-related equipment or perform any safety-function. Failure of this system could possibly require a plant shutdown, but would not create any conditions that would impact the ability to safely shutdown the plant and maintain it in a safe shutdown condition.

LDCN-FSAR-97-138 (SE 98-038)

FSAR Upgrade Project LDCN, Sections 9.5.2, "Communications System," and 9.5.3, "Plant Lighting System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with security communications.

Safety Evaluation Summary

It was concluded from the safety evaluation that revising the FSAR to indicate that the radio base stations utilized by the security communications centers have two-way communications channels rather than four two-way channels does not create a previously unanalyzed situation. Since the change does not actually modify the WNP-2 communications system, it does not create the possibility of changing it in a way that radio signals could interfere with plant equipment operation.

In addition, the communications centers control a total of five radio channels, so sufficient radio channel redundancy is maintained. There are two channels dedicated to operations and maintenance, while three channels are dedicated for security. This meets or exceeds the current FSAR commitment of two channels each. Therefore, no new plant operating modes, safety-related equipment lineups, accident scenarios, or equipment failure modes were identified.

LDCN-FSAR-97-138 (SE 98-039)

FSAR Upgrade Project LDCN, Sections 9.5.2, "Communications System," and 9.5.3, "Plant Lighting System." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with emergency signal tones.

Safety Evaluation Summary

It was concluded from the safety evaluation that changing the emergency signals for the General Evacuation and Containment Evacuation to the same tone as used for the Alert function does not create a previously unanalyzed situation. The public address announcement following the tone will identify actions to be taken and unique tones to identify these different functions is unnecessary. It was determined that the Emergency Plan was not impacted by this change. No new plant operating modes, safety-related equipment lineups, accident scenarios, or equipment failure modes were identified.

LDCN-FSAR-97-159 (SE 98-023)

FSAR Upgrade Project LDCN, Sections 13.1, "Organizational Structure," and 13.2, "Training." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the position of Shift Technical Advisor.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change does not result in any physical change to the configuration of the plant or have any impact on the method of operating plant systems.

Updating the FSAR to reflect current industry requirements and standards for training does not result in an unreviewed safety question. This change simply relocates (and updates) the Shift Technical Advisor (STA) commitments contained in Appendix B to Chapter 13 of the FSAR.

LDCN-FSAR-97-163 (SE 98-028)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the reactor building HVAC system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting the detailed description of the reactor building HVAC evaporative cooler control and instrumentation is not based on any physical change to the plant configuration or operating procedures, and is not associated with the safety-related engineered safety feature room coolers in the reactor building.

The reactor building evaporative cooler is associated only with the cooling function of the reactor building normal HVAC system. The temperature limits described in the FSAR are not impacted by this change. The specific setpoints and details of how the cooler works cannot be altered without appropriate evaluation of the change with respect to the requirements.

LDCN-FSAR-97-163 (SE 98-029)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with turbine sampling room cooling capacity.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting the description of the turbine sampling room air conditioning unit cooling capacity description does not impact any other structure, system or component. All of the features being deleted from the text are included on figure 9.4-6, except for the air conditioner cooling capacity. Deletion of the other information is considered an editorial clarification to consolidate information.

The specified function of the turbine building sample room air conditioning unit is to "provide tempered ventilation air to the sample room." This design function is not impacted by this change.

LDCN-FSAR-97-163 (SE 98-030)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the turbine building HVAC system.

Safety Evaluation Summary

It was concluded from the safety evaluation that eliminating the turbine building HVAC control instrumentation details that are not associated with any system functional requirements or related to any safety interfaces does not impact any other structure, system or component. The design functions of the system described in the FSAR are not impacted by this change.

The information being deleted is beyond the level of detail required to describe operation of the system. Any changes to the instrumentation and control information being deleted will not impact the capability of the system to perform its design function as long as the specified design functions in the FSAR are not impacted.

LDCN-FSAR-97-163 (SE 98-032)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the general service building HVAC system.

Safety Evaluation Summary

It was concluded from the safety evaluation that replacing the detailed description of the general service building HVAC system instrumentation with the statement, "Adequate instrumentation is provided to monitor and control operation of the system," does not impact the function of the system or any other structure, system or component.

The service building HVAC system instrumentation is only required to function to support system operation to maintain temperature for personnel comfort.

LDCN-FSAR-97-163 (SE 98-033)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the water treatment area and machine shop HVAC system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the deletion of descriptions pertaining to the water treatment area and machine shop HVAC system does not result in any physical change to the configuration of the plant or impact any safety-related or important to safety structure, system or component.

Malfunction or failure of this system will not impair normal or emergency operation of the plant and will not cause the release of radioactive materials. The system is non-safety related. A general description of the system design function and interfaces with other plant systems is retained.

LDCN-FSAR-97-163 (SE 98-034)

FSAR Upgrade Project LDCN, Section 9.4, "Heating, Ventilating, and Air Conditioning Systems." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the plant heating system.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting plant heating system condensate pump details such as a description of duplex type with a receiver tank, dual pumps, and float operated controls and revising the details associated with instrumentation monitoring does not result in an unreviewed safety question.

The plant heating system supports operation of nonessential HVAC systems and has no safety function. Failure of this system could potentially require a plant shutdown, but would not create any conditions that would impact the ability to shut down the plant and maintain it in a safe shutdown condition.

LDCN-FSAR-97-179 (SE 98-037)

FSAR Upgrade Project LDCN, Sections 9.5.4, "Diesel Generator Fuel Oil Storage and Transfer System," 9.5.5, "Diesel Generator Cooling Water System," 9.5.6, "Diesel Generator Starting Air System, 9.6.7, "Diesel Generator Lubrication System," 9.5.8, "Diesel Generator Combustion Air Intake and Exhaust System," and 9.5.9, "Plant Decontamination Facility." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the emergency diesel generator fuel oil transfer system.

<u>Safety Evaluation Summary</u>

This LDCN provided for: 1) changing the FSAR power supply for the fuel oil transfer pumps; 2) revising the specification for the fuel oil transfer pumps; and 3) changing the description of the operation of the fuel oil transfer pumps with respect to the day tank low level alarm and the amount of fuel available.

It was concluded from the safety evaluation that changing the FSAR description of the power source for the diesel fuel oil pumps is acceptable since the actual power source is diesel generator backed and, therefore, meets the intended safety function of supplying fuel to the diesel generators during the time period they require additional fuel (i.e., when they are in operation providing power when off-site power is unavailable).

In addition, it was determined that revising the FSAR to state that the fuel oil transfer pumps are sized to provide several times the maximum engine fuel oil consumption rate conveys the intent of installing over-capacity pumps. The change regarding the discussion of total fuel available in the day tank, along with the time remaining for corrective action to be taken when the low level alarm point is reached, provides for sufficient description of operation of the diesel fuel oil transfer system and its safety function.

LDCN-FSAR-97-179 (SE 98-045)

FSAR Upgrade Project LDCN, Sections 9.5.4, "Diesel Generator Fuel Oil Storage and Transfer System," 9.5.5, "Diesel Generator Cooling Water System," 9.5.6, "Diesel Generator Starting Air System, 9.6.7, "Diesel Generator Lubrication System," 9.5.8, "Diesel Generator Combustion Air Intake and Exhaust System," and 9.5.9, "Plant

Decontamination Facility." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the plant decontamination facility.

Safety Evaluation Summary

It was concluded from the safety evaluation that deleting specifics and details from the description of the plant decontamination facility does not result in any physical change to the plant configuration. The plant decontamination facility has no safety function and any malfunction or failure of the decontamination facility will not impact normal or emergency operations. A general description of the system design function is retained.

LDCN-FSAR-97-181 (SE 97-141)

This LDCN provided for a change to the FSAR to update and clarify retraining and replacement program requirements for the unit staff as part of continuing training. The use of ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," requirements with the added clarity of more recently developed INPO training guidelines, ensure that specific training objectives and criteria are used in preference to the more generic ANSI N18.1-1971 recommendations.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change provides a similar margin of safety by ensuring that specific training objectives and criteria are used. A detailed systematic approach to training method is employed at WNP-2 to analyze, design, develop, implement, evaluate and maintain unit staff training, requalification and replacement programs. Recommendations of ANSI N18.1-1971 are considered and incorporated into training programs as necessary.

However, all recommendations are not applicable to every unit staff position. Removal of the generic recommendation statement precludes confusion and potential conflict with the more detailed systematic approach to training by tailoring training requirements to job function.

LDCN-FSAR-98-007 (SE 98-007)

This LDCN provided for a change to the FSAR to add clarification regarding criticality monitors. Specifically, FSAR Question Q.331.026 and its response was incorporated into the text to clarify the application of criticality monitor alarm set points and associated calculational methodology.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change does not result in any physical change to the configuration of the plant or alteration in the method of operating plant systems.

Because WNP-2 is only licensed to handle critical mass quantities of slightly enriched material (fuel at $<$ 5.0 percent enrichment), the guidelines for alarm and response criteria are not applicable to fuel handling at WNP-2.

LDCN-FSAR-98-015 (SE 98-013)

This LDCN provided for a revision to the FSAR to delete reference to the service platform, service platform support and service platform slings. It was determined that using the platform results in unnecessary radiation exposure, and that tasks normally performed from the service platform can be accomplished at a lower dose from the refueling platform or the auxiliary work platform.

In addition, a change was made to indicate that the bypass plug is normally installed under all conditions. These changes were the result of the intent to remove the service platform from the plant.

Safety Evaluation Summary

It was concluded from the safety evaluation that removal of the service platform, platform support and associated slings does not constitute an unreviewed safety question. The platform, platform support and slings are not safety-related components and are not relied upon to mitigate the consequences of any anticipated operational occurrences or postulated accidents.

Operation of the refueling interlocks, which according to the FSAR are not essential for safety of the plant, will also not be impacted because a bypass plug for the service platform hoist load interlock will remain permanently installed.

LDCN-FSAR-98-018 (SE 98-014)

This LDCN provided for a change to the FSAR to reflect the surface treatments used on the control rod drive assemblies replacement components. The Standard Review Plan requires that the materials used in the control rod drive assemblies be reviewed for their susceptibility to stress corrosion cracking in reactor coolant, and that these materials be identified in the FSAR.

Safety Evaluation Summary

It was concluded from the safety evaluation that the material changes do not affect the safety function of the control rod drive assemblies or impact the plant in any way other than that discussed in the FSAR.

The materials were reviewed for their compatibility with the boiling water reactor coolant environment, as required by the Standard Review Plan, and no compatibility issues were identified that could increase the probability of an accident of malfunction of the control rod drive assemblies.

LDCN-FSAR-98-022 (SE 98-020)

This LDCN provided for a revision to the FSAR to reflect recent Health Physics department reorganization changes. In addition, a change was made to update the types and numbers of health physics instruments. A change was also made to revise the criteria which determines when an individual is required to obtain a whole body count.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes do not impact any important structure, system or component. The health physics program continues to provide the required detection capabilities and an adequate number of instruments is maintained to support normal and accident conditions. The revised criteria of determining when an individual must receive a whole body count remain conservative and continue to meet or exceed the regulatory requirements for monitoring internal radiation exposure.

LDCN-FSAR-98-040 (SE 98-022)

This LDCN provided for a revision to the FSAR to remove the option which allows the reactor core isolation cooling system to be shifted between the vessel injection mode and the test mode. The change will limit reactor core isolation cooling system operation to only the injection mode for normal system operation. This change does not apply to reactor core isolation cooling system surveillance testing when the system is declared inoperable and the Technical Specification action statements are entered. These changes were necessary because when operating in the test mode, valves RCIC-V-22 and RCIC-V-59 are not qualified to close against system pressure, and failure of these valves to close will prevent the system from performing its design basis vessel injection function.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change does not result in any physical change to the configuration of the plant.

System reliability is maintained and enhanced by this change. System operation to control water level can continue to be adequately accomplished using the manual or automatic mode and prevent unnecessary cycling of the system between Level-2 and Level-8.

LDCN-FSAR-98-046 (SE 98-036)

This LDCN provided for a change to the FSAR to provide for the option to demonstrate that, in lieu of re-solution heat treating, materials can be verified as unsensitized using means identified in Regulatory Guide 1.44, "Control of Sensitized Stainless Steel." Portions of the replacement emergency core cooling system suction strainers were stress relieved in a manner inconsistent with the FSAR. The alternative to re-solution heat treating is applicable to portions of engineered safety feature systems which are not part of the reactor coolant pressure boundary.

Safety Evaluation Summary

It was concluded from the safety evaluation that the stress relief of parts of the strainers for the "A" and "B" residual heat removal systems was found to have no adverse effect on the design, performance, or quality of the strainers. The testing of the strainer materials verified that the steel was not sensitized by the stress relief to the degree that would cause susceptibility to intergranular stress corrosion cracking.

The accept-as-is disposition was determined to have no impact on the operation of the engineered safety feature systems, nor their ability to perform their design functions to mitigate design basis accidents. The implementation of the evaluation processes results in no change to engineered safety feature materials, nor any degradation in the ability of the system to perform its design functions under all plant conditions.

LDCN-FSAR-98-061 (SE 98-052)

This LDCN provided for a revision to the FSAR to eliminate the specific value for when Operators initiate wetwell sprays. This is appropriate since the specific value for initiation of wetwell sprays was not used in the design analysis. In addition, the FSAR was changed to include Severe Accident Guidelines used during emergency conditions.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change is indicative of an event having already occurred requiring wetwell sprays, therefore it cannot impact the probability of the event occurring.

The changes to the FSAR do not result in an unreviewed safety question for any individual change and, therefore, no unreviewed safety question exists for the implementation of the Severe Accident Guidelines.

LDCN-FSAR-98-063 (SE 98-054)

This LDCN provides for a revision to the FSAR to accurately reflect existing and approved design features and administrative procedures that control access to safety system bypasses. These revisions to the FSAR provide a more complete description of the existing safety system bypasses and of the existing safety system bypass controls related to the use of keylock switches, access to keys, key location, and other approved methods for controlling such bypasses. The revisions do not alter existing plant designs or procedures.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The change does not result in any physical change to the configuration of the plant or impact or alteration in the method of operating plant systems.

Use of these controls is consistent with NRC recommendations and WNP-2 licensing commitments and does not result in an unreviewed safety question.

LDCN-FSAR-98-066 (SE 98-078)

This LDCN provided for a revision to the FSAR to reflect deletion of reference to the requirement for sampling for a degraded core condition by means of the post-accident sampling system within four hours after a postulated LOCA.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The change does not result in any physical change to the configuration of the plant or have any impact on the method of operating plant systems.

Removal of this information does not constitute an unreviewed safety question because the change is background information and does not affect the design, operation, or reliability of the service water system or the post-accident sampling system.

LDCN-FSAR-98-085 (SE 98-080)

FSAR Upgrade Project LDCN, Appendix F, "Fire Protection Evaluation." The safety evaluation portion of this LDCN pertained to clarification of the discussion associated with the fire protection program.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The change does not result in any physical change to the configuration of the plant or have any impact on the method of operating plant systems.

These administrative changes were made to accurately reflect the WNP-2 fire protection program, plant design features, and the post-fire safe shutdown program. The changes are consistent with Fire Protection License Condition 2.c (14).

LDCN-FSAR-98-087 (SE 98-087)

This LDCN provided for a change to remove from the FSAR the description of the Plant Support Facility laundry facility and its operation.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient.

Removal of the description of the Plant Support Facility laundry facility and its operation has no impact on the operation of WNP-2. This activity does not involve or affect any plant systems or equipment, does not impact radiological dose to the public, or adversely affect any margin of safety.

LDCN-FSAR-98-095 (SE 98-089)

This LDCN provided for a revision to the FSAR to reflect a revised flow for diesel cooling water system heat exchanger DCW-HX-1C. The minimum flow for DCW-HX-1C was increased from 780 gpm to 850 gpm.

Safety Evaluation Summary

It was concluded from the safety evaluation that increasing the flow for the heat exchanger ensures that the heat exchanger can remove its design heat load and maintain the diesel generator below its design operating temperatures. A temperature control valve on the diesel cooling water side of the heat exchanger ensures that increased flow through the service water system side does not result in the engine operating at less than minimum optimal temperature.

Flow balance procedures ensure that all other cooling loads receive adequate flow and are not adversely impacted by an increase in flow to the heat exchanger. The standby service water system is an accident mitigation system and cannot initiate an accident.

LDCN-FSAR-98-101 (SE 98-093)

This LDCN provided for a change to the FSAR to delete the public address system electronics failure alarm from the remote shutdown room. This alarm will continue to enunciate in the main control room and locally at the public address racks. The function of the alarm is to alert personnel to use alternate communication methods should the public address system fail under accident conditions. This change was required to accurately describe current plant configuration.

Safety Evaluation Summary

It was concluded from the safety evaluation that the plant address system is designed to be reliable, but is not required to accommodate a single failure within the system. The plant address system is one of the tools that could be utilized by the plant operator to communicate actions required to assist in plant shutdown following an event.

The primary purpose of the alarm is to identify system failure during normal plant operation so that action can be taken to restore operability and improve availability during an event. Personnel have other means, such as radio communications, to determine plant status.

LDCN-FSAR-98-111 (SE 95-040-01)

This LDCN provided for a change to the FSAR to reclassify non-safety related portions of the reactor water cleanup system from Quality Group C to Quality Group D pursuant to Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water- Steam- and Radioactive Waste-Containing Components at Nuclear Power Plants," Revision 3.

Safety Evaluation Summary

It was concluded from the safety evaluation that the code reclassification has no impact on the ability of the reactor water cleanup system, or any system, to perform its intended function. The reclassification also does not involve a special test or experiment, a change to the Technical Specifications, or represent any unreviewed safety question.

LDCN-LCS-98-043 (SE 98-027)

This LDCN provided for a change to Licensee Controlled Specification Table 1.3.5.1-1, "Emergency Core Cooling System Instrumentation Trip Setpoints." The emergency core cooling system actuation trip setpoint for the Drywell Pressure - High functions were changed from 1.65 psig to less than or equal to 1.68 psig.

Safety Evaluation Summary

It was concluded from the safety evaluation that the Technical Specification allowable values for these functions are less than or equal to 1.88 psig. The setpoint calculations for these instruments document that trip setpoints of less than or equal to 1.68 psig will continue to ensure that the Technical Specification allowable values are met.

The emergency core cooling system Drywell Pressure - High signals are not credited in any design basis accident or transient analyses. These trip setpoints do not impact the Technical Specification allowable values, which provide the margin of safety to the assumptions made in the accident analyses. Therefore, the changes to the trip setpoints do not impact the margin of safety of any Technical Specification.

LDCN-LCS-98-056 (SE 98-048)

This LDCN provided for a change to Licensee Controlled Specification Table 1.6.4.2-1, "Secondary Containment Ventilation System Automatic Isolation Valves," The allowable close-stroke time for reactor building ventilation supply isolation valves ROA-V-1 and ROA-V-2 was changed from ten seconds to 15 seconds.

Safety Evaluation Summary

It was concluded from the safety evaluation that changing the stroke time for these reactor building ventilation supply isolation valves has no impact on any previously analyzed accident or transient.

There are two limiting accidents for which credit is taken for secondary containment integrity. These are the loss of coolant accident and the fuel handling accident. Reactor building ventilation supply isolation valves ROA-V-1 and ROA-V-2 have an active safety function to close on a secondary containment isolation signal that, in combination with other accident mitigation systems, establishes a boundary for untreated fission products within the secondary containment structure.

Changing the isolation time from ten seconds to 15 seconds for the valves does not affect the radiological assessment performed for each of the design basis accidents which credit secondary containment integrity. The reason for this is that the time assumed for the unfiltered release and start of the standby gas treatment system is not exceeded. The five-second increase in isolation time is conservatively bounded by existing analyses.

LDCN-LCS-98-100 (SE 98-092)

This LDCN provided for a change to the required compensatory measure for Licensee Controlled Specification 1.7.2, "Control Room Emergency Chillers," to allow for the initiation of a Problem Evaluation Request, instead of submitting a special report to the Vice President, Nuclear Operations when the required compensatory measures and associated completion times are not met for the control room emergency chillers.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change does not result in any physical change to the configuration of the plant or alteration in the method of operating plant systems. The change only impacts the administrative response in the event that the control room emergency chillers are not restored to operable status within the required time frame as prescribed by LCS 1.7.2.

No credit for this administrative process is taken in response to any accident or transient. Furthermore, the administrative response does not have any direct impact on equipment important to safety or the method of operating such equipment.

Initiation of a problem evaluation request within 24 hours is a more appropriate action than submitting a special report to the Vice President, Nuclear Operations within ten days. Initiation of a Problem Evaluation Request within 24 hours provides a greater assurance of achieving timely operability of the control room emergency chillers.

LDCN-OQAPD-98-045 (SE 98-066)

This LDCN provided for a revision to the Operational Quality Assurance Program description to reflect several editorial changes and changes that were not considered a reduction in commitments. Changes were also made that were considered to be a reduction in commitments and included: 1) removal of the Quality department from in-line reviews of procurement documents; 2) Removal of the Quality department from in-line reviews of FSAR changes; 3) clarification of documents reviewed by the Quality department; and 4) removal of the function of planning/scheduling/outage from the Plant Operations Committee membership. A safety evaluation was performed to address those changes which were considered to be a reduction in commitments.

Safety Evaluation Summary

It was concluded from the safety evaluation that removal of in-line Quality department review of procurement documents, nonconformance documents and FSAR changes has no impact on any previously analyzed accident or transient. The Quality department will continue to perform reviews of these documents, but on a sampling basis, rather than the implied 100 percent review. The documents will continue to be reviewed for technical adequacy by qualified individuals prior to approval for release.

Removal of the function of planning/scheduling/outage from the Plant Operations Committee membership has no impact on committee function or charter. These changes are considered administrative in nature and do not have any direct affect on equipment important to safety or impact on the margin of safety as defined for any Technical Specification. Pursuant to 10 CFR 50.54, these changes require NRC approval prior to implementation.

LDCN-TSB-98-013 (SE 98-012)

This LDCN provided for a change to the Technical Specification Bases to clarify the temperature criteria used to determine acceptable operation of the preheaters associated with containment atmosphere control system recombiners. The reference to 550°F was removed as an upper temperature limit for containment atmosphere control system preheater performance during surveillance testing.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. Removal of 550°F as an upper temperature limit is not in conflict with the system operational description contained in the FSAR. Removal of this upper temperature limit restriction has no impact on the operation of the containment atmosphere control system.

The preheaters will adequately support the containment atmosphere control system function of recombining hydrogen with oxygen if the heaters are able to generate sufficient heat to achieve and maintain a catalyst bed inlet temperature of 500°F within 90 minutes of system startup.

Inlet temperature greater than approximately 500°F prevents degradation of the catalyst bed from halogens that could be present in the feed gas, and provides assurance that the gas possesses adequate activation energy to proceed with the recombination reaction.

LDCN-TSB-98-014 (SE 98-016)

This LDCN provided for a change to the Technical Specification Bases to reflect an improved test methodology for the excess flow check valves located in the instrument lines connected to the suppression pool. The methodology was revised to reflect that: 1) excess flow check valves in instrument lines which sense primary containment pressure will be tested at approximately 35 psig; and 2) excess flow check valves in instrument lines which sense reactor pressure will be tested at a pressure of 85 psig to 110 psig.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The activity is limited to changing test methodology to more accurately simulate accident-type pressures that are likely to be applied to excess flow check valves connected to the suppression pool.

The test pressure at which excess flow check valve testing will be performed does not impact the ability or functional purpose of an excess flow check valve to close, or increase the probability of isolating instruments important to safety. This activity is also bounded by existing instrument line break analyses.

LDCN-TSB-98-064 (SE 98-056)

This LDCN provided for a change to the Technical Specification Bases to reflect additional details pertaining to the emergency core cooling system LOCA time delay relay function and the initiating reactor Level 2 and drywell pressure instrumentation. The change is necessary to assure that the inoperability of the initiating instruments specifications contain appropriate branching to the time delay relay channel specification.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change adds descriptive information on the channel scope details to the LOCA time delay relay channel and the initiating instruments.

The change cannot increase the consequences of an accident since the change assures current Technical Specification operability requirements are met and the limiting condition for operation actions are implemented.

2.6.3 Miscellaneous Changes

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

Configuration Document Change Request CDCR 98-02-012 (SE 98-017)

This configuration document change request provided for a drawing revision to reflect the correct piping configuration associated with radioactive waste solids handling system valve PWR-V-619. The drawing was changed to show that the entrainment separator drains are routed from PWR-V-619 to dewatering pump PWR-P-4 rather than to the environment.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. This activity is a drawing change only. The configuration change does not impact any important to safety system or component.

Configuration Document Change Request CDCR 98-04-005 (SE 98-063)

This configuration document change request provided for the replacement of the setpoint values with process variables on all M560 logic diagrams. In addition, a note was added to each drawing which cross-references the associated instrument master data sheet for setpoint information.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. This activity is a drawing change only. Specific setpoint values are controlled through the preparation and revision of the setpoint calculations.

Setpoint calculations and associated instrument master data sheet revisions are controlled through approved processes and procedures. The change has no impact on the ability of any system to perform its intended safety function.

Configuration Document Change Request CDCR 98-07-010 (SE 98-082)

This configuration document change request provided for the revision to an FSAR drawing to reflect the correct logic inputs for high pressure core spray system level switches HPCS-LS-3A and HPCS-LS-3B and associated valve HPCS-V-15. Valve HPCS-V-15 is the high pressure core spray system suction from the suppression pool.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. This activity is a drawing change only. The logic circuits remain unchanged by this activity. This change has no impact on the function or operation of HPCS-V-15 or the high pressure core spray system.

Clearance Order 91-01-0108 (SE 98-044-00 and SE 98-0044-01)

This clearance order provided for reactor closed cooling water system temperature control valves RCC-V-TCV-71C and RCC-TCV-72A to be maintained in the full-open (failed) position until repairs can be made in the 1999 maintenance and refueling outage. Valve RCC-TCV-71C is a throttle valve located on the cooling water outlet of drywell cooling unit CRA-FC-2C. Valve RCC-TCV-72A is a throttle valve located on the cooling water outlet of drywell cooling unit CRA-FC-2A.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. Operation with RCC-TCV-71C and RCC-TCV-72A in the full-open position is within the design capacity of the reactor closed cooling water system. All five temperature control valves are normally full-open during power operation.

The flow of cooling water through the cooling coils of CRA-FC-2A and CRA-FC-2C will be maximized when the temperature control valves are completely open. This configuration will not affect the containment isolation function of the reactor closed cooling water system or its ability to remove heat from nonessential systems. Drywell temperatures will continue to be maintained within Technical Specification limits.

Emergency Plan Annual Review (SE 98-042)

The annual review of the WNP-2 Emergency Plan resulted in the update of: 1) several emergency response organization position descriptions; 2) examples provided for emergency action levels; 3) the description of protective action; 4) the description of the emergency dose projection system; and 5) floor plan descriptions for the joint information center and the operations support center.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes have no impact on any previously analyzed accident or transient and do not decrease the effectiveness of the WNP-2 Emergency Plan.

The changes made to the WNP-2 Emergency Plan enhance the ability of the Supply System to effectively protect the health and safety of employees and the public.

Plant Procedure PPM 1.3.1 (SE 98-055)

This procedure defines the key elements of the operating programs, policies and practices at WNP-2. The procedure was changed to reflect the increase in the number of required on-shift Health Physics technicians from two to three.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change would not adversely impact safe operation of the plant, degrade the previously evaluated margins of safety, or adversely affect structures, systems or components required to prevent or mitigate accidents or transients.

Revising the procedure to require three Health Physics technicians is in alignment with staffing requirements associated with the WNP-2 Emergency Plan and NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." This change aligns the procedure with regulatory requirements.

Plant Procedure PPM 2.4.2 (SE 98-059)
Plant Procedure PPM 2.4.3 (SE 98-059)

These procedures are the system operating procedures for the residual heat removal and low pressure core spray systems respectively. An evaluation was required to assess the impact of having the minimum flow control valves maintained in the closed position when the low pressure emergency core cooling systems are in the standby mode. The nuclear steam supply system supplier originally provided a design where the valves were in the open position during the standby mode.

Safety Evaluation Summary

It was concluded from the safety evaluation that maintaining the minimum flow control valves closed in the standby mode would not adversely impact safe operation of the plant, degrade the previously evaluated margins of safety, or adversely affect structures, systems or components required to prevent or mitigate accidents or transients.

When the various emergency core cooling system loops are in the standby mode, they will reliably respond to LOCA signals and be available to support the required safety function in accordance with the assumptions in the accident analyses.

Operation of these systems with the minimum flow control valves closed in standby does not adversely affect the safety function of the systems. The valves will open if a low-flow condition is present in a running pump.

Plant Procedure PPM 4.12.1.1 (SE 98-057)

This procedure provides direction for main control room evacuation and remote cooldown in the event the control room becomes uninhabitable. Discrepancies were identified in the original WNP-2- high impedance fault analysis that could result in the loss of fire safe shutdown power supplies to safe shutdown equipment due to fire-induced faults. The procedure was revised to include Operator actions and backup system status indication information to preclude loss of fire safe shutdown capability resulting from potential fire induced faults on unprotected circuits.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes would not adversely impact safe operation of the plant, degrade the previously evaluated margins of safety, or adversely affect structures, systems or components required to prevent or mitigate accidents or transients

These changes only impact post fire safe shutdown operating strategies. The changes do not affect safe plant operating conditions or accident or transient responses.

Plant Procedure PPM 4.12.4.1 (SE 98-058)

This procedure provides direction in the event of a fire in the plant. Discrepancies were identified in the original WNP-2- high impedance fault analysis that could result in the loss of fire safe shutdown power supplies to safe shutdown equipment due to fire-induced faults. The procedure was revised to include new Operator actions to preclude loss of fire safe shutdown capability resulting from potential fire induced faults on unprotected circuits.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes would not adversely impact safe operation of the plant, degrade the previously evaluated margins of safety, or adversely affect structures, systems or components required to prevent or mitigate accidents or transients.

These changes only impact post fire safe shutdown operating strategies. The changes do not affect safe plant operating conditions or accident or transient responses.

Problem Evaluation Request 298-0182 (SE 98-025-00 and SE 98-025-01)

This problem evaluation request documented a situation where it was discovered that stainless steels used for the screens of the replacement emergency core cooling system suction strainers had estimated yield strengths that exceeded FSAR limits (90,000 psi). The manufacturing process for the materials used for the strainer screens resulted in the cold-working of the screens. A safety evaluation was performed to determine the acceptability of the material.

Safety Evaluation Summary

It was concluded from the safety evaluation that the emergency core cooling system suction strainers have no role in the initiation of design basis accidents or transients. The strainers are passive, non-pressure retaining components that support the function of the emergency core cooling systems following a design basis accident or certain plant transients.

The use of cold-worked austenitic stainless steel in the replacement emergency core cooling system suction strainers has increased the susceptibility of the material to stress corrosion cracking. However, additional conditions are required to cause stress corrosion cracking (i.e., high temperature and adverse chemistry).

Those additional conditions do not exist, therefore, it was concluded that the inception of stress corrosion cracking in the strainer screens is not credible. It was determined that the strainers met all technical and functional requirements for an accept-as-is disposition.

Problem Evaluation Request 298-0499 (SE 98-049)

This problem evaluation request documented a situation where it was noted, during the outage jet pump inspection, that two of the auxiliary wedges installed in a previous outage were missing their respective handles. A safety evaluation was performed to assess the loose parts associated with the failure of the screws which attach the handles to the wedges, and to evaluate continued operation with existing wedge handles and the potential of a similar failure during subsequent cycles of operation.

Safety Evaluation Summary

It was concluded from the safety evaluation that there were no safety concerns associated with the postulated lost parts. The evaluation concluded that there is no potential for fuel bundle damage due to flow blockage, impairment of control rod function, or interference with the safety functions of the reactor water cleanup, residual heat removal, or control rod drive systems.

There is no potential for chemical or corrosion damage of vessel internal components. There is no increase in the probability of occurrence of a malfunction of equipment important to safety. The loss of the wedge handles does not challenge the integrity of the jet pumps.

Problem Evaluation Request 298-0523 (SE 98-050)

During previous cycles, set screw gaps were discovered on jet pumps and the corrective actions were to reset the inlet mixer and install wedges for those with gaps greater than 12 mils. This problem evaluation request was written to determine the impact of continued operation with gaps between the reactor recirculation system jet pump inlet mixer and associated restrainer bracket set screws.

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 could be conservatively maintained for at least six cycles, with unbounded vibration.

Therefore, it is conservative that operation for an additional two cycles with unlimited gaps will not challenge the riser braces and, therefore, will not challenge jet pump integrity.

Problem Evaluation Request 298-0782 (SE 98-069-00 and SE 98-069-01)

This problem evaluation request documented the flooding event due to rupture of fire protection system valve FP-V-29D. An evaluation was requested to address the impact on the fire protection system licensing basis with the temporary lineup of two fire pumps running continuously until appropriate actions can be taken to prevent system water hammer from the normal standby operating mode.

Safety Evaluation Summary

It was concluded from the safety evaluation that a period of up to four months of continuous running of the electric fire pumps, or two months of continuous running of the diesel fire pumps, could be tolerated without a reduction in reliability or pumping capacity below required values.

If permanent plant modifications to prevent recurrence were not complete within this period, an evaluation was to be completed to ensure that pump flow capacity has not degraded below limits. If the design limits were not challenged, the four or two month operating time could then be repeated.

Problem Evaluation Request 298-0797 (SE 98-071)

This problem evaluation request documented a situation where it was noted that the emergency core cooling system pump room doors are not leak tight in the reverse (non-hinge side) direction. The FSAR referred to the doors as being watertight. However, some leakage can be expected past the doors. An assessment was performed to address the leak tightness of the emergency core cooling system pump room doors. This was a follow-up activity as part of recovery efforts associated with the flooding event due to rupture of fire protection system valve FP-V-29D.

Safety Evaluation Summary

It was concluded from the safety evaluation that the situation does not: 1) introduce a new type of accident or transient different than any previously evaluated; 2) increase the severity of previously evaluated accidents or transients; or 3) increase the frequency of any previously evaluated accident or transient.

Installation of the pump room doors which leak when pressure is applied on the non-hinge side is acceptable. Leakage past a door potentially results in a malfunction of equipment in the room. The equipment in the emergency core cooling system pump rooms have previously been evaluated for water-induced malfunctions due to internal flooding.

The installation of these doors creates a potential flood path, however, the consequences are the same regardless of whether the flood originates from inside the pump room, from the drain lines or through the door.

Troubleshooting Plan 97004355 (SE 97-136)

This troubleshooting plan was developed to investigate the cause of erratic performance of the offgas condenser level control system. The plan was considered a high risk evolution because troubleshooting would be performed on in-service equipment that had the potential to cause an unexpected load reduction. Accordingly, a safety evaluation was requested by management.

Safety Evaluation Summary

It was concluded from the safety evaluation that the troubleshooting effort will not increase the probability or consequences of an accident. The activity will be performed on a system that is comprised of two independent systems or channels, each of which has the capacity to maintain proper main condenser level.

The troubleshooting plan does not: 1) introduce a new type of accident or transient different than any previously evaluated; 2) increase the severity of previously evaluated accidents or transients; or 3) increase the frequency of any previously evaluated accident or transient.

Work Order MBPO (SE 98-065)

This work order provided for the transport of residual heat removal system pump motor RHR-M-P/2C outside of the equipment removal aisle described in the FSAR. The pump motor had to be removed for repair as part of recovery efforts associated with the flooding event due to rupture of fire protection system valve FP-V-29D.

Safety Evaluation Summary

It was concluded that transportation of RHR-M-P/2C outside of the path described in the FSAR does not impact any previously analyzed accident or transient. Load drops that are not associated with lifts over irradiated fuel are not evaluated in the FSAR. However, the consequences are bounded by the instrument line pipe break.

The dose consequences at the site boundary are unchanged in the event of inadvertent contacting of instruments or instrument racks. The proposed load path will have no impact on any credited safety systems. Also, moving the travel path does not change the impact on structures due to potential load drop or deadweight considerations.

Work Order MVB3 (SE 98-086)

This work order provided for the installation of a freeze seal on a three-quarter-inch stainless steel process instrumentation line to allow for a weld repair. The instrument line taps into a suction elbow in the reactor recirculation system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no impact on any previously analyzed accident or transient. The application of a freeze seal to a line in the reactor coolant pressure boundary will not increase the probability of an accident or malfunction of equipment important to safety if the freeze seal failed. The failure of the freeze seal is bounded by existing analyses.

The implementation of the freeze seal allows for the repair of an existing leak and will result in restoration of pressure boundary integrity of the reactor recirculation system instrument line. The potential of brittle fracture of the reactor coolant pressure boundary is not a concern since the material is austenitic stainless steel. The ASME Code and 10 CFR 50 address brittle fracture and fracture toughness relative to ferritic materials and does not consider austenitic stainless steel to be susceptible to brittle fracture.

2.6.4 Tests and Experiments

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

Plant Procedure PPM 8.3.403 (SE 98-070)

This procedure provided for a test of the fire protection system to determine whether the fire protection water supply was susceptible to water hammer during the actuation of preaction valve 66 and a combined actuation of preaction valves 66 and 81. This test was a follow-up activity as part of recovery efforts associated with the flooding event due to rupture of fire protection system valve FP-V-29D.

Safety Evaluation Summary

It was concluded from the safety evaluation that the test would not adversely impact safe operation of the plant, degrade the previously evaluated margins of safety, or adversely affect structures, systems or components required to prevent or mitigate accidents or transients.

System enhancements and dynamic modeling of the fire protection system indicate that the system configuration is such that vulnerability to water hammer is minimized and system components will not be exposed to excessive loads.

The test provides input to a permanent resolution to a concern associated with fire protection system water hammer. The plant configuration associated with the test: 1) does not interface with plant safety-related equipment; 2) does not introduce new hazards; 3) does not reduce the ability of safety systems to perform during accident or transient conditions; 4) has no impact on the design, material or fabrication standards of equipment which could initiate accidents; and 5) does not change system interfaces.

2.7 Regulatory Commitment Changes (NEI Process)

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-105684-00 (Transient Combustible Control)

The original commitment description reads, "Special emphasis will continue to be placed on the limiting of combustibles in the Reactor Building, as well as throughout the plant. Applicable plant personnel (e.g., Operations, Maintenance and Health Physics departments) will be required to periodically review the applicable section of 1.3.10 (Plant Procedure 1.3.10, Fire Protection Program)."

The commitment was made in response to NRC Inspection Report 85-05. [Reference Letter GO2-85-206, dated April 17, 1985, GC Sorensen (SS) to JB Martin (NRC), "NRC Inspection Report 85-05."]

This commitment was deleted. Since the time-frame that the commitment was made, plant performance and process control initiatives in this area have improved such that the action is no longer necessary.

RCC-117070-00 (FSAR Drawings)

The original commitment description reads, "The GE drawings will be removed from the active design database by designating them as historical information, not to be updated. The FSAR will be changed to satisfy Regulatory Guide 1.70 by other means, so that use of the GE drawings will no longer be necessary."

The commitment was made in response to NRC Inspection Report 95-03. [Reference Letter GO2-95-135, dated July 17, 1995, JV Parrish (SS) to NRC, "NRC Inspection Report 95-03 - Reply to a Notice of Violation."]

This commitment was deleted. We are retracting the commitment and will maintain the General Electric (GE) process diagrams in the active design database and, thus, maintain them as required in the FSAR. The effort to remove information from the GE process diagrams that was still required by Regulatory Guide 1.70, and placing the information in some other location, was not considered to be cost beneficial over the continued maintenance of the GE drawings in the active database.

RCC-135731-00 (Transient Combustible Control)

The original commitment description reads, "Bi-weekly inspections will be conducted by a Fire Protection Engineer. (Violators will be required to provide responses.)."

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."]

This commitment was revised. Since the time-frame that the commitment was made, plant performance and process control initiatives in the area of transient combustible control have improved. Plant Procedure 1.3.10, "Fire Protection Program," also currently contains a management expectation for Fire Protection personnel to perform monthly tours.

Therefore, the commitment is revised to state that monthly inspections will be conducted by Fire Protection personnel. The statement pertaining to the requirement for violators to provide responses is deleted.

RCC-116677-00 (Radiation Exposure Control)

The original commitment description reads, "Plant Procedure 11.2.2.5, 'ALARA Job Planning and Reviews,' will be revised to reflect Radiation Protection Manager direction that, for non-routine high radiation area entries greater than 50 mrem per task, personnel are required to utilize the additional exposure controls contained in Attachment 3 of the procedure."

The commitment was made in response to NRC Inspection Report 95-16. [Reference Letter GO2-95-137, dated July 21, 1995, JV Parrish to NRC, "NRC Inspection Report 95-16 - Response to Apparent Violations."]

This commitment was revised to require implementation of the control of Plant Procedure 11.2.2.5, Attachment 7.3, "Additional Exposure Controls," for each non-routine entry into a high radiation area where an individual is expected to receive in excess of 50 mrem during that entry, and the duration of the job or task will be less than 24 hours.

The commitment revision preserves the Person-in-Charge (PIC) concept for those jobs which have the highest potential for exceeding anticipated exposure for tasks of short duration (the PIC is responsible for tracking job status against anticipated exposure). Tasks of longer duration (i.e., greater than 24 hours) are normally monitored by Health Physics and the Radiological Planning Group to ensure that job exposure is progressing as expected in relation to percentage of job completion. Unexpected exposures are quickly investigated and corrective measures implemented.

RCC-131354-01 (Scram Solenoid Pilot Valve Testing)

The original commitment description is, "A reference sample of SSPVs having exhaust diaphragms (consisting of 5% but not less than 5 rods) will be tested on a 60 day surveillance cycle. The 60 day surveillances will be performed until sufficient information is obtained."

This commitment was made in response to BWROG SSPV testing recommendations. [Reference Letter GO2-96-078, dated April 5, 1996, JV Parrish (SS) to AC Thadani (NRC), "CRD SSPVs with Viton Internals."]

This commitment was deleted. Sufficient data has been obtained to eliminate the 60-day augmented testing. Scram timing history results of the representative sample of 1995 vintage viton (Viton A) SSPVs indicates that the degradation of the viton diaphragms has plateaued. Therefore, the 60-day augmented testing is no longer required to ensure satisfactory control rod drive scram performance during the 120-day Technical Specification surveillance period.

Appendix A

Annual Personnel Radiation Exposure
Work and Job Function Report
Calendar Year 1998

WO[illegible] JOB FUNCTION REPORT
[illegible]LENDAR YEAR 1998

This report was produced with direct reading dosimeter data.

Number of Persons Receiving Over 100 millirem is 678

Total MAN-REM: 267.854

		Number of Individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contractors and Others
OPERATIONS &	Maintenance Personnel	60.13	1.45	115.39	11.381	0.331	52.229
	Operating Personnel	28.05	1.00	.99	10.743	0.116	0.382
	Health Physics Personnel	7.88	.29	.26	2.068	0.004	0.336
	Supervisory Personnel	6.18	1.16	.92	0.616	0.199	0.218
	Engineering Personnel	8.80	9.71	33.77	1.175	1.898	22.336
ROUTINE MAINTENANCE	Maintenance Personnel	57.07	1.91	121.59	32.693	0.987	40.375
	Operating Personnel	1.19	.00	.11	2.354	0.000	0.175
	Health Physics Personnel	17.79	.00	40.02	7.768	0.000	13.374
	Supervisory Personnel	2.38	.11	.12	1.027	0.031	0.050
	Engineering Personnel	4.11	3.33	11.59	0.887	1.594	2.370
INSERVICE INSPECTION	Maintenance Personnel	.90	.01	6.73	0.958	0.001	4.059
	Operating Personnel	.04	.00	.00	0.040	0.000	0.000
	Health Physics Personnel	.07	.00	.05	0.056	0.000	0.077
	Supervisory Personnel	.00	.00	.04	0.000	0.000	0.002
	Engineering Personnel	1.48	.71	.82	0.586	0.794	0.682
SPECIAL MAINTENANCE	Maintenance Personnel	11.06	.08	41.21	3.801	0.035	9.109
	Operating Personnel	.08	.00	.03	0.080	0.000	0.016
	Health Physics Personnel	1.13	.00	1.38	0.731	0.000	0.474
	Supervisory Personnel	.41	.10	.00	0.112	0.019	0.000
	Engineering Personnel	1.95	1.34	6.47	0.261	0.285	5.424
WASTE PROCESSING	Maintenance Personnel	.53	.57	.00	0.474	0.668	0.000
	Operating Personnel	.00	.00	.87	0.000	0.000	0.320
	Health Physics Personnel	.06	.00	.01	0.135	0.000	0.028
	Supervisory Personnel	.00	.00	.00	0.000	0.000	0.000
	Engineering Personnel	.00	.00	.00	0.000	0.000	0.000
REFUELING	Maintenance Personnel	21.30	.00	10.84	21.045	0.000	1.676
	Operating Personnel	.62	.00	.00	0.388	0.000	0.000
	Health Physics Personnel	1.11	.71	4.28	0.598	0.099	1.534
	Supervisory Personnel	3.09	.65	.91	1.687	0.269	0.242
	Engineering Personnel	1.68	5.76	11.36	0.395	0.944	1.972
TOTAL	Maintenance Personnel	150.99	4.02	295.76	70.352	2.022	107.448
	Operating Personnel	29.98	1.00	2.00	13.605	0.116	0.893
	Health Physics Personnel	28.04	1.00	46.00	11.356	0.103	15.823
	Supervisory Personnel	12.06	2.02	1.99	3.442	0.518	0.512
	Engineering Personnel	18.02	20.85	64.01	3.304	5.515	32.784
GRAND TOTAL		239.09	28.89	409.76	102.059	8.274	157.460

Appendix A
Annual Personnel Radiation Exposure
Work and Job Function Report
Description of Special Maintenance

- Installation of reactor closed cooling system filter demineralizer and coupon station
- Re-work of instrument rack tubing
- Modification of condensate system demineralizer COND-DM-1A septa and hold pump
- Resolution of Thermo-Lag fire retardant material issues
- Installation of fire-rated cabling in reactor and radwaste buildings
- Re-routing of feedwater system heater drainline piping
- Installation of turbine service water system isolation valves
- Replacement of emergency core cooling system suction strainers
- Removal of turbine service water system dead-leg piping to the offgas system vault HVAC units
- Re-work of biological shield wall penetration seals
- Modification of floor drain radioactive system sparger for tank FDR-TK-22
- Re-work of fire-rated penetration seals
- Removal of residual heat removal/condensate system cross-ties and installation of blind flanges
- Removal of undervessel/waste transfer detector
- Assessment of flood damage and repair of residual heat removal system pump RHR-P-2C
- Assessment of flood damage and repair of floor drain radioactive system level switch FDR-LS-2B
- Assessment of flood damage and repair of low pressure core spray system flow control valve LPCS-FCV-11
- Assessment of flood damage to electrical and instrument raceways and conduit
- Assessment of flood damage and re-work of penetration seals
- Installation of pipe cap on residual heat removal system radioactive equipment drain