ENERGY NORTHWEST
P.O. Box 968 ■ Richland, Washington 99352-0968

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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**
1999 ANNUAL OPERATING REPORT

The annual operating report for calendar year 1999 is attached. If you have any questions or desire additional information pertaining to this report, please contact either me or WA Kiel at (509) 377-4490.

Respectfully,

D.W. Coleman

DW Coleman
Manager, Regulatory Affairs
Mail Drop PE20

Attachment

cc: EW Merschoff, NRC - Region IV
JS Cushing, NRC - NRR
REIRS Project Manager, NRC – NRR
NRC Resident Inspector - 927N
DL Williams - BPA - 1399
TC Poindexter - Winston & Strawn

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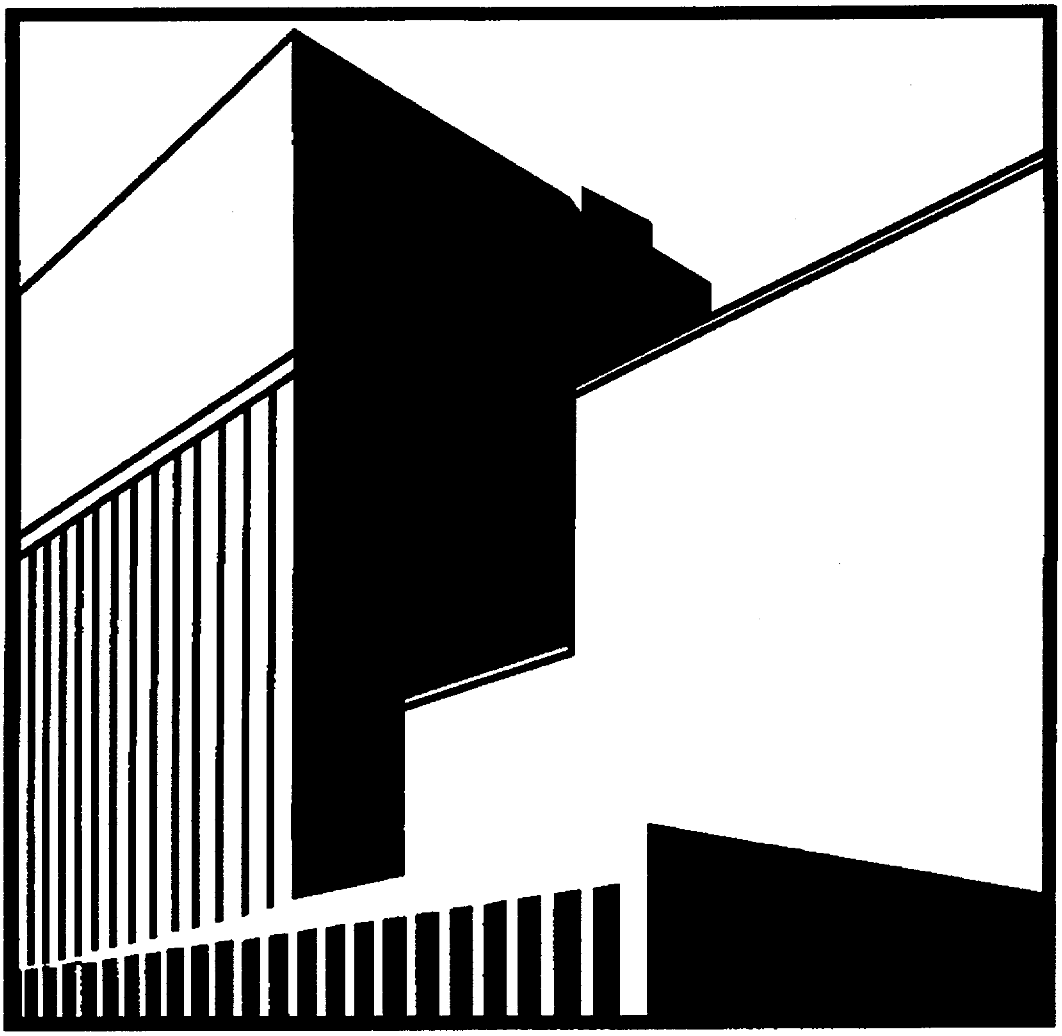

WNP-2 1999
ANNUAL OPERATING REPORT

ENERGY NORTHWEST

WNP-2

ANNUAL OPERATING REPORT

1999

DOCKET NO. 50-397

FACILITY OPERATING LICENSE NO. NPF-21

Energy Northwest
P.O. Box 968
Richland, Washington 99352

WNP-2 1999
Annual Operating Report

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1.0 Introduction

The 1999 Annual Operating Report for Energy Northwest's WNP-2 plant is submitted pursuant to Federal Regulations and Facility Operating License NPF-21. The plant is a 3486 MWt, BWR-5 that began commercial operation on December 13, 1984.

Operational Summary

On April 17, 1999, following an operational run of 106 days, the plant was shutdown for a fuel savings dispatch (non-outage reserve status) due to a planned transition from a 12-month to a 24-month fuel cycle. As part of this transition, the Spring 1999 refueling outage was moved to Fall 1999 to maximize the economic use of the reactor fuel and to support reload fuel design efforts for the extended fuel cycle.

Following the fuel savings dispatch outage, the plant was restarted on June 30, 1999. On July 13, 1999, the plant was returned to full power operation as planned and remained on line until September 17, 1999, when the unit was shutdown for the maintenance and refueling outage.

The maintenance and refueling outage officially ended on October 24, 1999, with synchronization to the electrical grid. The 36-day outage was the shortest in plant history. The plant was returned to full power operation on October 28, 1999, and with the exception of minor downpowers for rod pattern adjustments, remained at or near that power level for the remainder of the year.

Outage Summary

The fuel savings dispatch and the R-14 maintenance and refueling outages were successfully completed during 1999. Significant planned and emergent activities included:

- Modifications associated with resolution of post-fire safe-shutdown Thermo-Lag raceway fire barrier issues.
- Installation of an oscillation power range monitor (trip capabilities are currently deactivated).
- Six-year overhaul of emergency diesel generators E-DG-1 and E-DG-2.
- Refurbishment of circulating water system pump CW-P-1C.
- Inspection of several ABB fuel subassemblies for proper orientation.
- Replacement of 248 fuel assemblies.

- Disassembly, inspection, and repair of residual heat removal system pump RHR-P-2B.
- Replacement of 125VDC battery E-B1-7.
- Replacement of high-pressure core spray system 125VDC distribution panel DP-S1-HPCS.
- Replacement of ten main steam relief valves.
- Inspection of reactor feedwater spargers.
- Inspection, removal, testing and replacement of several snubbers.
- Tube cleaning and eddy current testing of several heat exchangers.
- Arc-spray coating of cross-under piping between the high-pressure turbine and the moisture separator reheater.

<u>Other Highlights</u>

Significant resources were expended on Year 2000 (Y2K) readiness during 1999. The project consisted of detailed assessments, quality assurance reviews, software and systems replacements and upgrades, testing and validation, and contingency planning. As a result of these efforts, no problems were identified during the Y2K rollover period.

Extensive efforts were undertaken in the resolution of post-fire, safe-shutdown Thermo-Lag raceway fire barrier issues. The efforts included revisions to safe-shutdown analyses, revisions to fire protection evaluations, and physical plant modifications. The resolution efforts were completed in 1999 as scheduled and consistent with the confirmatory order to the WNP-2 Operating License.

Preparations were made during the year for transition of the plant from a 12-month to a 24-month fuel cycle.

Records were set at WNP-2 for low radiation exposure during the year. For example, the total radiological dose for 1999 was 158 rem, which was the lowest since the second year of commercial operation in 1985.

During 1999 new power generation records were set for the months of November and December. The net electrical generation was 822,540 megawatt-hours and 852,055 megawatt-hours respectively.

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 NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes."

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 that 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 99-04, Guidelines for Managing NRC Commitment Changes* 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 1999

- At the beginning of the month, the plant was at full power operation. On January 25, 1999, power was reduced to 70 percent to repair a failed power supply for the adjustable speed drive (ASD) channel ASD-1B/1. The reason for the failure was due to a design flaw in the control board.

- On January 26, 1999, the plant returned to full power operation and remained at or near that level for the remainder of the month.

February 1999

- At the beginning of the month the plant was at full power operation. The plant remained at that level for the remainder of the month excluding downpowers for purposes of economic dispatch load following at the request of Bonneville Power Administration (BPA).

March 1999

- At the beginning of the month the plant was at full power operation. On March 29, 1999, power was reduced from 88 to 70 percent to repair a failed power supply for ASD channel ASD-1A/1. (The plant had been at 88 percent power for economic dispatch (load following) at the request of the Bonneville Power Administration.) The reason for the failure was due to a design flaw in the control board. On March 30, 1999, the plant was returned to full power operation.

April 1999

- At the beginning of the month the plant was at full power operation. On April 17, 1999, the main generator was removed from service and the plant was shutdown for the fuel savings dispatch outage.

May 1999

- The plant was in the fuel savings dispatch outage for the entire month.

June 1999

- At the request of BPA, the end of the fuel savings dispatch outage was extended on June 24, 1999. The plant remained in the fuel savings dispatch outage for the entire month.

July 1999

- The fuel savings dispatch outage officially ended on July 4, 1999, with synchronization to the electrical grid. The plant returned to full power operation on July 13, 1999.
- On July 14, 1999, all available control rods were fully withdrawn and the plant entered final feedwater temperature reduction.

August 1999

- At the beginning of the month the plant was at 93 percent power and coasted down to 79 percent at the end of the month.

September 1999

- At the beginning of the month the plant was at 79 percent power and coasted down to 72 percent power on September 16, 1999.
- On September 17, 1999, the main generator was removed from service and the plant was shutdown for the R-14 maintenance and refueling outage.

October 1999

- The R-14 maintenance and refueling outage officially ended on October 24, 1999, with synchronization to the electrical grid. The plant returned to full power operation on October 28, 1999.
- On October 29, 1999, power was reduced to 83 percent for rod sequence exchange. The plant returned to full power operation on October 30, 1999.

November 1999

- At the beginning of the month, the plant was at full power operation. On November 5, 1999, power was reduced to 85 percent for control rod pattern adjustments and performance of the main turbine trip functional test. On November 6, 1999, the plant returned to full power operation and remained at or near that level for the remainder of the month.

December 1999

- At the beginning of the month, the plant was at full power operation. Power was reduced to 80 percent in conformance with the regional Year 2000 contingency plan on December 31, 1999.

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

Containment atmosphere control system heat exchanger CAC-EV-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.

Control Rod Drive System

Twenty-three accumulators were refurbished which included replacement of elastomers and limit switches.

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

Emergency Diesel Generator System

The six-year overhaul was performed on emergency diesel generators E-DG-1 and E-DG-2.

Emergency diesel generator system heat exchangers DCW-HX-1B1 and DCW-HX-1B2 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.

High-Pressure Core Spray System

Filter replacements and turbocharger inspections were performed for high-pressure core spray system diesel generator HPCS-DG-3. In addition, the air start valves were rebuilt.

Main Steam System

Ten safety relief valves were refurbished as part of an ongoing corrective maintenance program to minimize leakage into the wetwell.

Nondestructive examination (eddy current) inspection sampling was performed on main steam system moisture separator/reheater heat exchangers MSR-HX-1A-A2 and MS-HX-1B-C2 tubing as part of an ongoing heater and heat exchanger tube cleaning and maintenance program.

Miscellaneous Systems

Inspection and testing were performed on inverter E-IN-2 following the occurrence of several voltage swings. As a result of the troubleshooting efforts, a circuit card and isolation transformer were replaced.

Residual Heat Removal System

Residual heat removal system pump RHR-P-2B was disassembled and inspected. A one-piece shaft sleeve nut was installed. Pump casing welds were also inspected.

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

Pinion gears were replaced on the motor-operator for residual heat removal system valve RHR-V-8.

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 Specification 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 1999 portion of Cycle 14. There was also no evidence of failed fuel for the calendar year 1999 portion of Cycle 15.

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. If we had identified any significant errors, they would be reported pursuant to 10 CFR 50.72 and 10 CFR 50.73 and are not included in this report (although we did not report any).

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 1999, there were no changes in the ECCS LOCA evaluation model or application of the model for the SPC fuel. However, an input error pertaining to the counter current flow limiting coefficients was identified in the application of the SAFER analysis for Siemens 9x9 fuel. The SAFER code models counter current flow limiting coefficients in the upper part of the bundle at the upper tie plate. For the Siemens 9x9 fuel, the upper tie plate flow area has been enlarged to reduce the pressure drop for the higher-powered bundles. As a result, the counter current flow limiting is expected to occur at the upper spacer. The error resulted in a +15 degrees Fahrenheit impact on the Peak Cladding Temperature (PCT).

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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

For the ABB fuel, there were no errors in an ECCS LOCA analysis model or application of the model for 1999. However, one change relating to a cladding dimension deviation was identified in the application of the ECCS LOCA analysis model. The change resulted in a +4 degrees Fahrenheit impact on the PCT.

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

This section contains summaries of the safety evaluations (SE) completed for activities implemented during 1999 and is included pursuant to 10 CFR 50.59.

Regulation 10 CFR 50.59 and Energy Northwest 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 that 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 existing safety evaluation, 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 91-0190-0 (SE 98-098)

This modification provided for the installation of equipment and a hard-wired jumper across the oscillation power range monitor (OPRM) – reactor protection system (RPS) contacts that cause the OPRM system to only monitor reactor instabilities.

Safety Evaluation Summary

It was concluded from the safety evaluation that this design change does not affect the operation of the sub-systems of the power range neutron monitoring system. None of the accidents or equipment malfunctions affected by these sub-systems are affected by the presence or operation of the OPRM. The existing average power range monitor (APRM), local power range monitor (LPRM), and RPS systems still perform their intended functions. The new equipment has been designed and tested to the electronic magnetic interference (EMI) requirements, which ensure correct operation of the existing equipment. The new system is designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The new OPRM equipment is designed and installed to the same system requirements as the existing equipment and has no impact on the existing functions of the APRM system.

PMR 92-0276-0 (SE 99-0067)

This design change provided engineering analysis to complete the Thermo-Lag reduction plan which is needed to meet our 10 CFR 50.54 commitments to the NRC in regard to Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers." This change does not involve any fieldwork.

The specific changes made by this design change are as follows:

- Abandoning Thermo-Lag raceways and the 20 feet non-combustible zone. Abandoning small length of 3M Interam raceway fire barrier.

- Switching post fire safe shutdown (PFSS) divisions for fire areas TG-1 and RC-2.

- PFSS analysis changes including the elimination of the "dedicated" fire area methodology, revised spurious signal analysis, revised high impedance fault methodology and analysis, classification of PFSS components, cable ampacity and voltage drop calculations, and miscellaneous items from the revised PFSS analysis.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes do not constitute an unreviewed safety question. Abandoning reliance on Thermo-Lag 330-1 for raceway wraps is warranted by the NRC (Generic Letter 92-08) and the Nuclear Utility Management Resource Council (NUMARC)/Nuclear Energy Institute (NEI) fire testing which have shown that Thermo-Lag does not provide the required hourly fire rating. Existing plant drawings, standards, calculations and procedures are in place to control abandoned Thermo-Lag 330-1. This change provides the engineering analysis and the fire protection evaluations to ensure that the existing cables within the abandoned Thermo-Lag raceway barriers, when subjected to a design basis fire, do not prevent PFSS.

The required changes to fire abnormal operating procedures ensure PFSS is achievable in worst-case fire conditions. The revised PFSS plan, together with the previously approved Thermo-Lag modifications, meet all design requirements and WNP-2 commitments relative to fire protection design and nuclear safety.

PMR 92-0276-2 (SE 97-068)

This design change provided for the installation of the intermediary raceways and cables between electrical enclosures for the elimination of all credited PFSS Thermo-Lag. This modification also provided for a reduction in the number of new raceways that require protection in the cable chase, reactor building, and the cable spreading room.

Safety Evaluation Summary

It was concluded from the safety evaluation that the newly installed cables are not connected to any active plant system. Connection of the cables is to be accomplished by separate design changes. New raceways and cables added by this modification are installed to Quality Class 1 requirements. The potential impact to safety due to installation activities is accounted for in existing plant procedures. In addition, the increase in plant combustibles in various plant fire areas and the addition of loads to new and existing Seismic Class 1 raceway supports are within previously analyzed limits.

PMR 92-0276-2 (SE 97-068-1)

The revised design change reduced the width of the Division 1 PFSS trays in the diesel generator corridor to accommodate future Darmatt (fire barrier) installation on the trays.

Safety Evaluation Summary

It was concluded from the safety evaluation that the replacement of existing trays with narrower trays in the diesel generator corridor affects certain cable ampacities and tray load supports. These revised variables are within analyzed limits and, therefore, do not affect the plant safety analysis. Only Division 1 power and control cabling is affected by the reduction in tray width. Within this area of tray reduction, there are no cable terminations or splices and the cables are securely fastened by the entrance and exit penetration. Therefore, accidents resulting from unplanned cable pullouts are not credible.

PMR 92-0276-4 (SE 99-0008)

This design change provides for the following three changes:

- The installation of Darmatt KM-1 and 3M Interam raceway fire barriers.
- The relocation of the process sampling radioactive (PSR) system line to allow installation of Darmatt. During relocation, the post-accident sampling system (PASS) valves PSR-V-146 and 147, and containment isolation valve PSR-V-145, were closed disabling the PASS wetwell atmosphere sample capability.
- The installation of thermal barrier (Kaowool) to service water line SW(46)-2-1 to prevent MCC Room 410 from overheating due to room cooler operation during design basis fire conditions.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes provide raceway fire barriers that have the NRC-required fire rating and meet flamespread, combustibility, seismic, chemical composition, voltage drop, and ampacity derating criteria. The Kaowool wrap ensures the service water fluid temperature supplying the motor control center room cooler does not exceed PFSS design limits. These changes involve passive plant fire barriers only and do not affect: any other plant equipment; the ability of safety systems to perform their functions during accident conditions; design, material or construction standards of equipment which could initiate accidents; system interfaces; or the means of initiation of any safety analysis report accidents. In addition, no new fire hazards are introduced as a result of this design change.

PMR 92-0276-4 (SE 99-0008-1)

This revised design change substituted "*many*" for "*47*" in the response to Safety Evaluation 99-0008. The specific sentence that was changed follows.

> The installation of 3M Interam consists of wrapping a four-foot length which has Thermo-Lag, adding 2 layers to *many* supports and having a Darmatt KM-1 interface at each end of the conduit run.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is no safety significance from changing the number of supports from 47 to many.

PMR 92-0276-4 (SE 99-0008-2)

The revised design change identified additional circuits that require protection from the effects of fire. This change provides protection in the form of rerouting circuits out of the fire area. The components involved are the diesel generator E-DG-1 fuel oil transfer pump and the transfer pump room exhaust fan. The modification only routes cables through wall penetrations into existing protected raceway and installs conduits. This modification does not provide the final determs and reterms of the power and control circuits, nor direct any post-modification testing activities.

Safety Evaluation Summary

It was concluded from the safety evaluation that rerouting the Division 1 diesel generator fuel oil transfer pump through the adjacent Division 3 high-pressure core spray (HPCS) diesel generator room does not result in any electrical separation or missile impacts. Rerouting the circuits through the secondary containment boundary was performed using approved plant procedures breaching secondary containment and installing appropriate penetration seals, as required, to ensure secondary containment integrity.

PMR 92-0276-6 (SE 98-095)

This modification provided for the installation of intermediate raceways, cables, and other alternative PFSS design features for the protection of Appendix R instrument cables that route through fire areas R-1, RC-2, and RC-3. Alternative PFSS design features include the use of Whittaker fire-rated cable assemblies and rerouting circuit cabling through fire areas that do not require protection. This design change provided the electrical connections to these alternatively protected cables. This design change does not provide for any process connection changes, instrument functional changes, or instrument setpoint changes.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change does not alter any instrument function or setpoint. No process connections are affected by this change. Cable terminations were completed under Quality Class 1 installation requirements using approved plant procedures. These instruments provide a passive indication function only and contain no functional or electrical interfaces with any other important to safety structure, system, or component.

PMR 92-0276-8 (SE 98-100)

This modification provided for the installation of intermediate raceways, cables, and other alternative PFSS design features for the protection of Appendix R instrument cables that route through fire areas R-1, RC-2, and RC-3. The change disconnects existing PFSS component circuits and other associated circuits and reconnects them to the new cables installed by design change 92-0276-2. In addition, a fuse was added to the supervisory circuit for SCI-RLY-

OCS2/1/2 in a planned field change request, in order to meet requirements of high impedance fault analysis. Implementation of these changes helps eliminate WNP-2's dependence on Thermo-Lag as a credited Appendix R fire barrier material.

Safety Evaluation Summary

It was concluded from the safety evaluation that an unreviewed safety question does not exist as a result of this design change. The changes to cables and associated equipment were performed during plant shutdown conditions in accordance with approved plant procedures. In addition, the cable terminations and cable routes meet all plant seismic, electrical separation, ampacity, and voltage drop requirements. This design change results in fewer fire protection barriers being needed to ensure PFSS circuits function properly when required. No system functional or operational changes were made by this design change. The affected circuitry functions in the same manner as before the modification.

PMR 92-0276-9 (SE 98-103)

This design change provided for cable routing, disconnection, reconnection, and testing of the PFSS circuits including the relocation of motor control center components and the deactivation of motor-operated valves. These changes are in support of the resolution of PFSS Thermo-Lag raceway fire barrier issues. This design change involved several components and sub-components in the PFSS system.

Safety Evaluation Summary

It was concluded from the safety evaluation that no system functional or operational changes result from this design change. The only exceptions are RHR-V-47A and B which were made manually operated valves locked in the open position. However, these valves do not perform an active safety function. All changes in circuit parameters for the remaining components were evaluated and found to be within plant design and margins.

PMR 92-0276-10 (SE 98-084)

This design change provided for cable routing, disconnecting, reconnecting and testing of several PFSS circuits. The new cables and necessary raceways that are used for this activity were installed by design change 92-0276-2 as part of the Thermo-Lag reduction program.

Safety Evaluation Summary

It was concluded from the safety evaluation that no unreviewed safety question exists as a result of this change. The cables and associated equipment were worked (routed, disconnected, reconnected, and tested) during plant shutdown, in Mode 4 or 5, in accordance with approved plant procedures. The cable terminations were inspected and the circuits tested to ensure compliance with plant requirements. Specialty areas including environmental, pipe whip and missile control system failure analysis, flooding, penetrations, seismic, and spraying and wetting have been reviewed for these cables and terminations and found to meet WNP-2 design requirements.

Following completion of the work specified in this design change, the circuitry functions in the same manner as before the modification. However, due to the consolidation of circuits and new cable routing, fewer raceway fire barriers are needed to ensure PFSS circuits function properly when required.

No system or operational changes were made by this design change. All revised circuit parameters related to the rerouting of cables were evaluated and found to be within plant design margins as previously evaluated in the FSAR.

PMR 92-0276-11 (SE 98-077)

This modification provided for fuse installation, disconnection, reconnection, and testing of PFSS circuits to help eliminate the use of Thermo-Lag for the PFSS circuit protection.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that the cable terminations and fuse installations were implemented in plant shutdown conditions in accordance with plant procedures. In addition, the cable terminations and fuse installations meet all plant seismic, electrical separation, ampacity, and voltage drop requirements. Circuits or systems removed from service adhere to licensing basis document requirements with required redundant systems available and were worked within applicable time limits. No system functional or operational changes were made by this design change. All revised circuit parameters were evaluated and found to be within plant design margins.

PMR 92-0276-13 (SE 99-0049)

This modification provided for the reduction of WNP-2's reliance on Thermo-Lag as a PFSS fire barrier. Circuits were rerouted through compatible fire areas or raceway fire barriers that provide the fire resistance to ensure PFSS capability is maintained.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that these changes are in accordance with fire protection License Condition 2.c (14) and meet the requirements for fire-rated barriers, electrical separation, pipe breaks and missiles, flame spread, voltage drop, ampacity derating, seismic capacity and environmental qualification. In addition, the rerouting of these circuits has been reviewed for electrical separation and missile/pipe break impacts with none identified. Post-modification testing was performed after the newly routed cables and circuits were reconnected and functional. All work associated with this design change was performed during plant shutdown conditions and installation was performed in accordance with existing approved plant procedures and instructions.

PMR 93-0046-0 (SE 97-090)

This modification provided for hardware and software modifications to: 1) place the plant data information system (PDIS) functions on the local area network (LAN); 2) make the PDIS functions available by means of the LAN to the graphic display system, the workstations in the main control room, and the technical support center; and 3) permanently deactivate the Prime Computer and spare it in place.

Safety Evaluation Summary

It was concluded from the safety evaluation that the accident/transient analysis, equipment malfunctions, and margin of safety are not affected by this modification. Installation in accordance with the FSAR ensures that isolation is maintained between 1E and non-1E systems, structures or components. Analysis of AC power load requirements concluded that the availability of power for safety-related functions would not be affected. The heat load from new components has been evaluated and determined to have no adverse affect on the ability of the control room cooling system to maintain the temperature within 104 degrees Fahrenheit as specified in the Technical Specifications. In addition, new installation complies with seismic 2 over 1 requirements. This design change does not change the function of the PDIS or the LAN. In addition, installation, testing, and operation does not affect the functions of any safety-related systems or equipment.

PMR 94-0007-1 (SE 99-0034)

This modification provides for the replacement of battery E-B1-7, a non-safety-related, balance of plant, 125 VDC battery. Replacement of the battery is necessary because of a design deficiency in the post-lock post-seal which results in cracked post-nut covers and cell top covers. The battery manufacturer redesigned the post-seal to eliminate the design deficiency. In addition, to provide additional design margin for battery E-B1-7, this modification replaces the existing type 2GN-13 cell units with higher capacity type 2GN-15 cell units.

Safety Evaluation Summary

It was concluded from the safety evaluation that no unreviewed safety question exists as a result of these changes. The design change replaces the non-safety-related battery (E-B1-7) with a battery of the same form, fit, and function as the existing battery. The replacement battery is made by the same manufacturer, is of the same type, and is the same physical size as the existing battery cells. The replacement battery performs the same function as the existing battery, that is, provides a reliable source of power to its connected non-safety-related loads. The larger capacity of the new battery allows the connected loads to run for a longer period of time before reaching a fully discharged state. The larger capacity changes three interface parameters: 1) additional weight; 2) increased hydrogen generation; and 3) increased fault current. The battery rack and floor are adequate to support the heavier cell units and retain its Seismic Category 1M capability during a seismic event. The existing heating, ventillating, and air-conditioning (HVAC) flow was determined to be sufficient to prevent the buildup of hydrogen to an explosive concentration. In the event of an HVAC failure, plant procedures direct that temporary ventilation be established to prevent the buildup of hydrogen to an explosive concentration. The available short circuit fault current generated by the new larger capacity battery has been evaluated and determined to be within the rating of the bus distribution panel, fuses, and cables. Although the DC bus remained energized during installation and testing of the replacement battery, loss of this non-safety-related DC bus would result in de-energizing only non-safety-related loads that are addressed in plant procedures. Replacement of E-B1-7 was performed during shutdown conditions to avoid any potential plant system transients.

PMR 94-0043-3 (SE 97-140)

This design change provided for the modification of RHR-V-42C to prevent pressure locking by installing a ½-inch diameter external bypass line from the valve body cavity area to the connected piping on the containment side of the valve. This bypass line vents pressure between the wedge discs to the high-pressure side of the valve. This design change does not add a new component to the design of the plant. The ½" diameter piping is considered a "passive" component.

Safety Evaluation Summary

It was concluded from the safety evaluation that the presence of the ½-inch bypass line does not interfere with the performance of the residual heat removal (RHR) valve or RHR system function. The containment isolation function is not affected since only the non-containment side disc seals when the highest pressure is on the containment side of the valve. The bypassing of a single valve disc will have no affect on the isolation function of the valve. The non-containment side valve disc continues to provide the containment isolation barrier. The ability of the valve to isolate, as required by the Technical Specifications, is not affected by this modification. This modification increases the reliability of valve RHR-V-42C to open on demand and enhances reliability of the low-pressure coolant injection function of RHR.

PMR 97-0040-2 (SE 98-062)

This modification provided for implementation of the WNP-2 Cycle 15 core design. In addition, the ultimate heat sink (UHS) analyses was revised to incorporate a 24-month fuel cycle, a full spent fuel pool, and a greater siphon differential.

Safety Evaluation Summary

It was concluded from the safety evaluation that assumptions and bounding conditions were identified and accounted for relevant to the cycle 15 safety analysis. The COLR limits are established to maintain the probabilities and consequences in the safety analysis and maintain margin of safety. The enrichment change in the reload fuel is within the approved scope of NRC approved methodology. In addition, the design change does not affect the safety function of the UHS or standby service water (SSW) systems. The safety function of these systems is to provide cooling water for up to 30 days without makeup water to help mitigate the consequence of a transient. The UHS and SSW systems continue to meet their basic design requirements.

PMR 97-0042-0 (SE 99-0009)

This modification provided for the permanent deactivation of the radwaste sludge tank level instrumentation. The deactivated components were spared in place due to as low as reasonably achievable (ALARA) dose and cost considerations.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes do not result in a degradation of nuclear safety or an unsafe configuration. The modification meets all design requirements and commitments for WNP-2 as described in the licensing basis documents.

The affected instrumentation has no connections to Class 1 electrical power and no functional interfaces with Class 1 or 2+ components. The only function of the instrumentation is to provide local and remote indication of sludge level. Tanks EDR-TK-1, FDR-TK-22, RWCU-TK-104A and B, and CPR-TK-92A and B are non-safety-related components, and the associated systems have no safety-related functions. In addition, the deactivated instrumentation has no direct physical connection with Quality Class 1 piping. Therefore, this modification does not affect any Quality Class 1 components, systems, or structures.

PMR 98-0043-0 (SE 98-107)

This modification provided for the replacement of the existing stack radiation monitoring system. The stack monitoring computers continued to exhibit memory file errors, slow and unreliable communication links between computers, and the computers and their software programs were not Y2K compliant.

This modification upgraded the stack monitor computers, incorporated the ethernet-based links to the electronics rack, and installed "Omnigam" spectrum analysis software. In addition, appropriate components were reclassified from Quality Class 2+ to Quality Class 2.

Safety Evaluation Summary

It was concluded from the safety evaluation that the stack monitoring system is not safety-related and is non-seismic with the exception of components located in the main control room that must meet seismic 2 over 1 requirements. The new computers, software, and peripherals perform the same basic function as those previously installed. In addition, the new hardware and software is

Y2K compliant. Fuse isolation has been provided to assure that failure of these components does not adversely affect the Quality Class 2+ portion of the instrument loop.
The system function remains unchanged. The system is a monitoring system only and is not safety-related. It monitors specific processes and provides information to plant operating personnel for use in mitigating the consequences of an accident. It has no control function that could affect any safety-related system or component.

PMR 98-0070-0 (SE 98-091)

This modification provided for the installation of redundant vacuum breakers on selected standpipes in the reactor, radwaste, and turbine buildings to prevent water hammer in the fire protection system. In addition, a check valve on the 2-½ inch pipe section of RWB-1 standpipe was installed, hanger FP-HGR-114 on the RWB-2 standpipe was modified, and a time delay relay with a 10-second setting was installed on the FP-P-110 controller.

Safety Evaluation Summary

It was concluded from this safety evaluation that the addition of the vacuum breaker valves does not prevent the fire protection system from performing its safety function. An analysis of this change in conjunction with stress analysis of the individual standpipes/hangers with the highest transient loads, concluded that the plant configuration prevents the stresses in the system from reaching unsafe levels with the worst anticipated transient, assuming one vacuum breaker is operable on each standpipe.

PMR 98-0070-1 (SE 99-0013)

This design change provided for the implementation of remaining actions necessary for long-term resolution of fire protection water system (FPWS) deficiencies highlighted by the June 1998 water hammer and flooding event.

Specifically, this design change provided for: 1) modifications to primary and secondary FPWS pumps and controllers; 2) removal of local alarms from the radwaste, turbine, and diesel generator buildings; 3) replacement of D64 Grinnel brand deluge valve with a standard Reliable brand deluge valve; and 4) justification for the permanent use of cast steel valves for FP-V-29D and FP-V-394.

Safety Evaluation Summary

It was concluded from the safety evaluation that each of the changes are being implemented to improve the reliability and safety of the fire protection system. None of the changes adversely impact PFSS. These changes meet fire protection License Condition 2.c (14) and do not constitute an unreviewed safety question.

PMR 98-0082-0 (SE 99-0017)

This modification provided for an improved seating configuration for the main steam safety relief valves (MSRVs) utilizing a redesigned disc insert. Crosby Valve and Gage Company, the original designer and manufacturer of SRVs, developed the improved seating configuration. This new seating configuration is necessary because the MSRVs had excessive seat leakage during plant operation.

This design change provided for the modification of all the MSRVs, 18 installed and 10 spare, to incorporate the vendor's improved seat design to reduce seat leakage. Modified and improved MSRVs were installed in the plant in accordance with approved procedures.

This design change is only partially plant implemented. The existing 10 spare MSRVs were modified prior to the R-14 refueling outage and 10 MSRVs were modified during R-14. The implementation of this design change will remain open until the completion of the installation process in the future.

Safety Evaluation Summary

It was concluded from the safety evaluation that based on the prototype testing and the evaluation of the changes by Crosby Valve and Gage Company and WNP-2 Engineering, the modification does not impact the MSRVs operation, performance, or function, except for seat leakage reduction. The MSRVs continue to maintain their rated capacity, setpoint, and blowdown requirements. All the MSRVs safety setpoints as listed in the Technical Specifications remain unchanged; therefore, the proposed modification does not reduce the margin of safety as defined in the basis for any Technical Specification. The proposed modification does not create a new accident or any new malfunction of important to safety equipment.

PMR 99-0030-0 (SE 99-0050)

This modification provided for the replacement of the microwave transfer trip (MWTT) analog radio UHF communication system. This change involved only the installations of field-routed conduits and fiber optic communication links between WNP-2 and the Ashe Substation. The purpose of this modification was to increase the reliability of the MWTT communication system.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change does not result in a degradation of nuclear safety or an unsafe configuration. The modification activities meet all design requirements and commitments for WNP-2 as described in the licensing basis documents. In addition, the implementing activities described in the design change meet the requirements of applicable WNP-2 procedures. Therefore, the safe operation of the plant is not challenged.

The conduits and cables were installed to meet Quality Class 2 and Seismic Category 2 requirements. The new penetration C503-1001 meets Seismic Category 1M requirements. The fiber optic cables meet Institute of Electrical and Electronics Engineers (IEEE) 383 Flame Test Requirements and are plenum-rated in accordance with the FSAR. Therefore, the added combustible loading is minimal and does not present a fire hazard.

PMR 99-0044-0 (SE 99-0022)

This modification provided for the relocation of the reactor water cleanup (RWCU) pump seal purge line from downstream of the secondary containment bypass leakage valves to upstream of control rod drive (CRD) system check valves CRD-V-524 and 525. This provided for the elimination of an unanalyzed secondary containment bypass leakage path from the control rod drives back through the RWCU pump seal into the RWCU filter/demineralizer lines. This modification installed new valves, piping, and a filter station. The new filter station eliminated any debris that is loosened up as a result of flushing the carbon steel line.

Safety Evaluation Summary

It was concluded from the safety evaluation that there is no increased risk of a reactor scram or other trips caused by the installation. The hangers, piping, tubing, and equipment have been analyzed as Seismic 1 and installed as Seismic 1M. The relocation of the RWCU pump seal purge line does not interfere with the performance of the RWCU or CRD systems. This modification eliminated a secondary containment bypass leakage path should the CRD pumps become inoperable following a design basis accident.

PMR 99-0066-0 (SE 99-0025)

This modification provided for resizing of the following components:

125 Volt DC Main Distribution Panel E-DP-S1/1 (Division 1)

- The replacement of the 200 amp fuse with a 125 amp fuse in cubicle 2E for protection of cable 1D11-008.

- Increased the size of the feeder cable conductor 1D11-003 from 1/0 gauge cable (capable of carrying 91 amps) to 4/0 gauge cable (capable of carrying 166 amps). This change corrects the inadequate overload protection for feeder cable 1D11-003 without compromising E-MC-S1/1D loads.

125 Volt DC Main Distribution Panel E-DP-S1/2 (Division 2)

- The replacement of the 200 amp fuse with a 110 amp fuse in cubicle 2C for protection of cable 2D12-004.

- Increased the size of the feeder cable conductor 2D12-009 from 1/0 gauge cable (capable of carrying 91 amps) to 4/0 gauge cable (capable of carrying 166 amps). This change corrects the inadequate overload protection for feeder cable 2D12-009 without compromising E-MC-S1/2D loads.

Safety Evaluation Summary

It was concluded from the safety evaluation that the replacement of existing electrical components with electrically similar components does not constitute an unreviewed safety question. The replacement of electrical components having similar failure modes that comply with electrical and safety requirements, and meet the criteria of Quality Class 1 and Seismic 1 requirements does not constitute an unreviewed safety question.

The installation of the new components described by this design change was performed during a mode when the respective divisions of 125VDC power are not required.

PMR 99-0081-0 (SE 99-0046)

This modification provided for the improvement of the fault protection and coordination within the Division 3 (HPCS) 125VDC power system, and protects the HPCS battery from damage by replacing the existing main distribution panel, which employs molded case breakers, with new panelboard using properly coordinated fusible switch units. The existing breaker is located within the Diesel Generator Control Panel. Since there is not room for the replacement panel to be mounted in the space vacated by the old breaker panel, the new fusible panelboard is mounted nearby on a freestanding support. Power and control conduits were installed from the new panelboard to the existing cable tray system. The main feeder cables from the battery and the battery charger and branch circuits cables to bus SM-4 and instrument rack IR-P028 were de-termed, pulled back into the tray, and routed to the replacement panelboard. The remaining seven branch circuit cables were spliced to like cable types and extended from the diesel generator control panel to the new panelboard. Work was not performed in Modes 1, 2, or 3 and was completed using existing plant procedures and applying Technical Specifications requirements.

Safety Evaluation Summary

It was determined from the safety evaluation that this change does not constitute an unreviewed safety question. Improving the fault protection and coordination within the HPCS, 125VDC power system, by replacing its molded case circuit breakers with fusible switches, is not a complex change and does not change the safety function or failure modes of the HPCS system.

TER 98-0004-0 (SE 99-0001)

This modification allows the chemical feed pumps CF-P-41 and CF-P-42 to reliably inject the CF-TK-41 and CF-TK-42 contents into the plant service water system as per their design intent. The existing components CF-P-41, 42, CF-M-P/41, 42, CF-RV-821, 822, CF-FG-41, 42, CF-PI-41 and 42 were replaced with components having higher working pressure ratings. Valves CF-V-830 and 836 were no longer needed and these equipment piece numbers were deleted. Feed pump CF-CP-1 was altered to provide AC power to the new control units CF-CONT-41 and 42 which will power the replacement variable speed DC pump motors.

Safety Evaluation Summary

It was concluded from the safety evaluation that this modification does not have the potential to adversely affect any previously evaluated accident or equipment malfunction, does not create the possibility of a different type of accident or malfunction, and does not affect the Technical Specifications Bases. In addition, the installation and required post-maintenance testing activities do not impact any engineered safety function or impair any credited barrier. Changes made by this modification do not constitute an unreviewed safety question.

TER 98-0016-0 (SE 99-0003)

This modification provided for the replacement of ½-inch cooling jacket water (CJW) valve CJW-V-750 with a 1½-inch valve. CJW-V-750 provides a usable alternate emergency fuel pool cooling water discharge flow path associated with emergency cooling of the control air system (CAS) compressors and dryers on loss of CJW and/or loss of turbine building service water (TSW) condition.

Safety Evaluation Summary

It was concluded from the safety evaluation that the normal functional operation of the CJW and TSW systems is not affected by this change. In addition, the emergency fuel pool cooling operation of these systems is not impacted by this modification. The affected CJW system piping and associated components remain non-safety-related, Quality Class 2, Seismic Category 2, and non-radiologically contaminated. The availability of the CAS system is also not adversely impacted by this change.

TER 98-0101-0 (SE 98-104)

This modification provided for the replacement of the spring and spring steps in low-pressure core spray (LPCS) relief valve LPCS-RV-18. In addition, this design change increased the setpoint from 427 to 442 psig.

Safety Evaluation Summary

It was concluded from the safety evaluation that the replacement of the spring and spring steps and the increase of the setpoint of LPCS-RV-18 does not constitute an unreviewed safety question. Valve LPCS-RV-18 continues to perform its safety function. The increase in the setpoint of LPCS-RV-18 from 427 to 442 psig is within the 10 percent of system design pressure tolerance allowed by calculation, thus, the pressure boundary of the LPCS system is still adequately protected. The spring and spring step replacement provides a spring with the proper range. The spring material changes from cadmium plated carbon to stainless steel, and is the replacement piece part provided by the vendor. The spring material change does not affect any safety function of LPCS-RV-18. Valve LPCS-RV-18 functions as before only at a slightly higher pressure.

TER 98-0119-0 (SE 99-0055)

This modification provided for the relocation of the hot bleedoff backpressure differential controller COND-DPIC-1A pressure sensing line connections from the pump outboard hot bleedoff line to the pump inboard/outboard common header. This new location enables regulating the hot bleedoff system backpressure using the average hot bleedoff pressure from both ends of the pump instead of just one end (outboard). This results in a more accurate averaged hot bleedoff pressure, allowing better reactor feedwater pump seal water pressure regulation.

Safety Evaluation Summary

It was concluded from the safety evaluation that the relocated pressure taps continue to perform the same intended function as the current design. Installation occurred when the system is not required to be operable. The design change does not result in a change to the plant that adversely impacts any safety-related structure, system, or component.

TER 98-0139-0 (SE 99-0060)

This modification provided for the installation of ½-inch connection, isolation valve, and compound pressure gauge with a calibration port onto the 4-inch loop seal drain line of steam jet air ejector (SJAE) condensers COND-HX-8A and 8B. The pressure gauge assembly was located on the vertical drain line directly beneath the associated SJAE condenser in the respective SJAE rooms at turbine building elevation 441 feet.

Safety Evaluation Summary

It was concluded from the safety evaluation that installation of the proposed pressure gauge assembly onto the existing loop seal drain line of the SJAE condensers COND-HX-8A and 8B does not affect the safety-related systems or functions. In addition, the design change does not affect the design function or operation of the air removal, heater drain, or condensate systems.

The proposed pressure gauge assembly and existing interfacing piping and equipment are classified as non-safety-related, Quality Class 2, Seismic Category 2 components. This change, in conjunction with existing drain line temperature monitoring capability, results in greater assurance that the necessary water loop seal required to prevent bypass leakage of non-condensibles and loss of main condenser vacuum is maintained.

TER 99-0006-0 (SE 99-0002)

This modification provided for the addition of a manually operated ¾ inch ball pattern vent valve to the TSW system during plant outage conditions. Valve TSW-V-786 was located upstream of TSW-V-21 on a common discharge header off the main turbine oil coolers. The ball valve that was added has a working pressure rating greater than that of the TSW system piping.

Safety Evaluation Summary

It was concluded from the safety evaluation that the affected TSW system piping, valves, and associated components remain non-safety-related, Quality Class 2 and Seismic Category 2. In addition, the TSW system remains non-radiologically contaminated. This design change does not change the functional operation of the TSW system.

TER 99-0145-0 (SE 98-105, 98-105-1, and 98-105-2)

This modification, which was prompted by operating experience at the Limerick station, provided for the upgrade of the suction heads of the emergency core cooling system (ECCS) Ingersoll-Rand pumps. This involves the following:

- The suction head material was upgraded from cast iron to cast steel.
- The suction head was increased to six webs instead of three webs.
- For the HPCS and LPCS pumps, the suction head bearing material was upgraded from bearing bronze to graphalloy.
- A new one-piece shaft sleeve/nut replaces the separate suction head shaft sleeve and shaft nut and eliminates the current sleeve key and retaining ring.

Safety Evaluation Summary

It was concluded from the safety evaluation that the upgrades of the suction heads of the ECCS pumps do not change their capabilities to perform their safety functions. All of the changes result in improvements in the reliability of the individual pumps. The cast steel results in significantly higher allowable stresses; it is a tougher, more ductile material. The increase to six webs reduces the stresses in each web. In addition, the one-piece design for the shaft sleeve/nut eliminates the axial play, prevents pressure pulsations from acting on the end surfaces, and is held by the shaft threads.

TER 99-0158-0 (SE 99-0061)

This modification provided for the repair of the reactor recirculation (RRC) instrument line that provides input to the APRM flow units APRM-FU-43 and 23 by means of flow transmitters RRC-FT-24C and D respectively. Repair was necessary as the line was discovered leaking at the ¾-inch socket weld connection to the 24-inch RRC 'B' loop elbow on the suction side of RRC-V-23B.

Safety Evaluation Summary

It was concluded from the safety evaluation that the repair of the RRC instrument line using a modified tee fitting and temporary plug during Mode 4 does not result in a degradation of nuclear safety or result in an unsafe configuration. The new installation meets all design requirements and commitments for WNP-2 as described in the licensing basis documents. In addition, implementation of the new installation results in improved reliability for the affected RRC instrument line.

TMR 99-0004 (SE 99-0021)

This modification provided for an interim action to continue plant operation with FD-RIS-1, 2, 3 and 4 deactivated.

In 1992, contamination was discovered in the storm drain system. As a result of that discovery, a decision was made to prevent the direct discharge of turbine building sumps T1, T2, T3, and the berm area around the condensate storage tank to the storm drain system. Drainage from these areas was isolated from the storm drain system and is now routinely directed to the liquid radwaste system. At about the same time, the circuit breakers for the motor operators for drain valves FD-V-10, 15, 18, and 24 were locked open preventing their operation. These valves close to isolate the sump outputs to the storm drain system, and when locked closed, permanently divert sump effluent flow to the radwaste system. Therefore, the automatic function to accomplish this diversion on high activity as detected by radiation monitors FD-RIS-1, 2, and 3 has been eliminated.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed activity permanently diverts the non-radioactive floor drain sumps T1, T2, T3 and the CST berm area effluents to the radwaste system rather than divert them only when radioactivity is detected. The effluent is processed as radwaste and released through the normal monitored release. The level of radioactivity in the non-radioactive floor drain sump effluents must meet the limits placed on all radwaste discharges. Therefore, this modification does not increase the consequences of a malfunction of equipment important to safety. In addition, since this change permanently replaces the system in its design function condition, it does not increase the probability of a malfunction of equipment important to safety previously evaluated in the licensing basis documents.

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.

LDCN-FSAR-98-009 (SE 98-008)

This LDCN provided for a change to the FSAR description of the new fuel receipt and inspection process. Specifically, corrections were made to inaccurate statements regarding: 1) the control of combustible material in the new fuel vault; 2) uses of the general purpose grapple; 3) use of a temporary metal cover in the new fuel vault; and 4) the amount of water shielding provided by the fuel preparation machine when handling irradiated fuel assemblies.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes to the FSAR are being made to reflect the way current plant procedures direct and control the process of receiving new fuel. The current procedures apply lessons learned making the process more efficient. In addition, the processes directed by current plant procedures do not result in a reduction of the margin of safety from the way the fuel receipt process was originally described in the FSAR. The changes are not specific to any mode and do not increase the probability of an accident previously analyzed by the safety analysis report.

LDCN-FSAR-98-142 (SE 99-0043)

This LDCN provided for a change to the FSAR to correct an error in FSAR Section 9.5.4.5, "Instrumentation Requirements," and Figure 9.5-1.4, "Diesel Oil and Miscellaneous Systems," which incorrectly stated that the control switches for the three diesel fuel oil storage tank transfer pumps are located in the main control room. These three control switches are located on their respective diesel generator local control panels and are part of the diesel fuel oil transfer subsystem.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change has no safety significance since it has no impact on the evaluated accidents or malfunctions. Each diesel generator has an independent fuel oil supply and the loss of one diesel generator has been assumed in the transient and accident analysis. In addition, the malfunction described in FSAR section 9.5.4.2, "System Description" can be mitigated by the local control switch.

LDCN-FSAR-98-145 (SE 99-0006)

This LDCN revised the FSAR to change the location for conducting internal dosimetry and respirator fit testing from the Plant Support Facility to WNP-2. In addition, the locker change room areas were eliminated from the Health Physics facilities description. The size and location of the described locker change rooms were not sufficient to meet operating and outage needs. Temporary change areas described in the FSAR, located near the work location, can exclusively provide this function.

Safety Evaluation Summary

It was concluded from the safety evaluation that a change in the physical location of the equipment used to perform internal dosimetry and respirator fit testing activities does not constitute an unreviewed safety question as these activities are not included in the basis for any accident. In addition, the new location of the equipment does not place it near safety-related or important to safety equipment.

LDCN-FSAR-98-147 (SE 98-102)

This LDCN provided for the removal of references to a spare 125 volt DC battery charger and a spare 250 volt DC charger from Chapter 8.3 of the FSAR.

In 1977, Energy Northwest requested that the WNP-2 Architect/Engineer evaluate the feasibility of a scheme which would allow a minimum transfer time for connecting the spare 125 volt DC battery charger to its respective bus. The technical evaluation proposed several alternative designs. The recommended scheme required the installation of two spare chargers, one for each division, with special modifications to allow for ease of connection in the event of failure of the on-line unit. It was decided that it was not necessary to replace one 125 Volt DC charger with another in less than two hours as this was not required to complete construction, would not improve the plant completion date, and was not necessary for licensing. Both the spare 125 Volt DC charger and a spare 250 Volt DC charger were placed in spare parts inventory.

The original FSAR, section 8.3.2.1.1.1, "125 Volt DC Division 1 and 2 Systems," contained the following statement: "A spare unconnected charger of the same rating is provided." A similar statement was made in section 8.3.2.1.4, "250 Volt DC (Division 1) System." SAR Change Notice SCN 92-094 revised the above two statements, as an editorial change to read as follows, "A spare charger of the same rating is available for replacement." Deletion of these statements from the FSAR removes the commitment to maintain spare chargers.

Safety Evaluation Summary

It was concluded from the safety evaluation that the DC power system battery chargers are not an initiator of accidents. The probability of an event occurring is completely independent of the availability of spare parts. Station blackout (SBO) coping time is four hours without chargers. In addition, SBO does not assume a concurrent single failure or a design basis accident.

The determination of what parts to keep in spare inventory is based strictly on economics, not safety; therefore, removal of the commitment to maintain a spare charger does not adversely affect the safe operation of WNP-2.

LDCN-FSAR-99-016 (SE 99-0011)

This LDCN provided clarification regarding how the separator is moved with respect to reactor cavity water level. The change is a result of an evaluation request to calculate the clearance between the separator and the reactor pool components. This was to determine if the separator could be moved completely underwater, as the FSAR appeared to imply. The results of the evaluation indicated that with the currently maintained water level in the spent fuel pool/cavity (605 feet 7-1/2 inches) or with nominal water level (605 feet 10-1/3 inches) that the top of the separator would be between three and seven inches out of the water. Therefore, this change was implemented to revise the FSAR to indicate that the separator is not moved while being completely covered by water.

Safety Evaluation Summary

It was concluded from the safety evaluation that moving the separator to and from the dryer-separator pit with the top eight to 12 inches of the separator out of water has no safety significance and minimal radiological consequences when moved during Mode 5. Raising the top of the separator out of water rather than moving it completely under water has no affect on evaluated accidents or malfunctions. No safety functions are affected by this change and no barriers are impaired by this change. In addition, no post-modification testing is required as a result of this change.

LDCN-FSAR-99-022 (SE 99-0014)

This LDCN provided for a change to FSAR Table 13.1-1, "Minimum Shift Crew Composition," and PPM 1.3.1, section 4.13, "Shift Complement," to reflect the following:

1. An Improved Standard Technical Specification NRC Safety Evaluation allowing an SRO Licensed Shift Technical Advisor (STA) to assume the control room command function.

2. The allowance for the Shift Manager to be absent from the site for a period of time not to exceed two hours. This is the requirement for the Operations required shift crew composition. Previously, the Shift Manager was excluded from this requirement in the FSAR.

Safety Evaluation Summary

It was concluded from the safety evaluation that the use of a SRO licensed STA in the control room command function was specifically approved by the NRC in the safety evaluation for the Improved Technical Specifications (ITS). Specifically, the staff stated, "The ITS changes this provision to allow the STA to fulfill the control room command function provided the individual has an active SRO license. The CTS excluded the STA because the STA was formerly not a member of the Operations department. Since the CTS requirement was approved, the Operations department assumed responsibility for the STA position. Therefore, the STA is appropriately qualified to fulfill the control room command function provided the STA holds an active SRO license. This change conforms to the STS format and is acceptable." The use of an SRO licensed STA in the control room command function ensures that a qualified individual maintains the control room command function.

The allowance for the unexpected absence of the Shift Manager from the site is consistent with the allowance provided for other required Operations shift crew members. This two hour allowance provides time to locate a qualified Shift Manager and for that individual to get to the site and take the shift. This allowance is specifically provided in Technical Specification 5.2.2, "Unit Staff" and 10 CFR 50.54(m)(2)(i), Note 1.

Since the proposed FSAR changes have already been reviewed and approved by the NRC as documented in the safety evaluation report associated with Technical Specification Amendment 149 (ITS), these changes cannot, by definition, constitute unreviewed safety questions.

LDCN-FSAR-99-028 (SE 99-0028)

This LDCN provided for a change to FSAR sections 9.4.13, "Service Building," 9.4.14, "Water Treatment Area and Machine Shop," and the removal of associated figures and tables from the FSAR.

Safety Evaluation Summary

It was concluded from the safety evaluation that the affected sections of the FSAR describe the HVAC systems serving the general services building and the machine shop areas. These systems and equipment are Quality Class G and provide for personnel comfort cooling only. These FSAR sections are not required by Regulatory Guide 1.70, Revision 2, for the FSAR nor is there any requirement in NEI 98-03 for their inclusion in the FSAR. Changes to these sections of the FSAR eliminate unnecessary descriptions and burdensome document maintenance requirements. Control of modifications to these systems is not compromised by changes to these sections. System changes and modifications are still covered by approved procedures and are not affected by the change. Changes to the interfaces between these systems and the power block are controlled by PPM 1.4.1, "Plant Modifications," so that power block integrity is maintained. These systems and their FSAR descriptions do not contribute to normal, abnormal, or emergency operation of the plant. They are not required for safe shutdown nor are they required to maintain the plant in a safe condition. Specifically, in the case of the operations support center (OSC), the service building HVAC is not needed as a condition for manning the center. If the service building lunchroom environment became unbearable because of a failure of the HVAC system, the OSC would change locations. Changing these sections of the FSAR does not constitute an unreviewed safety question.

LDCN-FSAR-99-044 (SE 99-0035)

This LDCN provided for a change to FSAR sections 7.5.2.2.3, "Regulatory Guide Conformance" and 1.8.3, "Balance of Plant Scope of Supply Evaluation." Specifically, the change takes exception to the acceptance of the Regulatory Guide 1.97 recommendation to use portable multi-channel gamma spectrometers for release assessment and analysis.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change does not constitute an unreviewed safety question. Regulatory Guide 1.97, Revision 2, Table 2, "Plant and Environs Radioactivity," recommends the use of portable multi-channel gamma spectrometers for release assessment and analysis. WNP-2 uses air sampling techniques and practical analysis methods of silver zeolite air sample cartridges to detect fission products released to the environment. These techniques are preferable to the use of portable multi-channel gamma spectrometers because emergency field teams are able to quickly obtain a sample and analyze it away from the plume exposure hazard. In addition, portable multi-channel gamma spectrometers require power supply and temperature stability in order to ensure that instrument calibration is maintained and sample analysis results are accurate. These conditions are not obtainable in the field.

Analysis of environmental samples obtained by emergency field teams is performed as a recovery action and not during the plume phase of an accident. Capability also exists at the emergency operations facility to perform detailed analysis of field team environmental samples.

LDCN-FSAR-99-045 (SE 99-0033)

This LDCN provided for the removal of the commitment to vent the non-seismic portion of the CRD system to secondary containment during Type A containment leakage rate testing. This change does not constitute a change to the plant. This commitment is currently described in Note 4 of Table 6.2-16, "Primary Containment Isolation Valves." Note 4 is also modified to explicitly state the CRD lines are exempt from Type A and Type C testing requirements. The design information currently in Note 4 is being relocated to more appropriate sections of the FSAR.

Safety Evaluation Summary

It was concluded from the safety evaluation that the original FSAR submitted to the NRC during the Operating License review process did not include the commitment to vent the non-seismic portion of the CRD system to secondary containment during Type A testing. The commitment to vent was added in February 1981 by Amendment 13 as a response to NRC Question 022.010. This question required identification of containment isolation valves that would not be subjected to Type C leak testing and the justification for not performing the tests.

At approximately the same time WNP-2 was responding to the NRC question, the industry was in the process of responding to various issues that arose from Three Mile Island. One of the issues was the design of the CRD insert and withdrawal lines. General Electric, the Boiling Water Reactor Owners Group, and the NRC were evaluating the CRD system design and had prepared several reports that investigated the design attributes from a containment isolation perspective and conformance to the GDC.

The NRC evaluation of the studies that were performed is contained in NUREG-0803, "Generic Safety Evaluation Report Regarding the Integrity of BWR Scram System Piping," dated August 1981. The report concluded that the design attributes of the CRD system were such that although the design departs from the explicit requirements of GDC 55, the departure is justified on other defined bases and therefore meets the requirements of GDC 55.

Elimination of the commitment to vent the non-seismic portions of the CRD system during Type A testing is supported by the requirements of 10 CFR 50 Appendix J as amplified by additional guidance from Regulatory Guide 1.163, ANSI/ANS-56.8-1981 and 1994. These documents state that Type A and Type C testing is not required for systems that do not have a pathway open to primary containment atmosphere during a design basis accident. The CRD withdrawal and insert lines do not provide such a pathway.

No accidents or anticipated operational occurrences initiated by the CRD insert and withdrawal lines or hydraulic control units are described in the FSAR. No new accidents or failure modes were identified. The consequences of accidents evaluated remains unchanged. No changes are being made to the CRD system or processes that support reliable operation. The CRD system and CRD mechanisms continue to perform their safety function to support reactor shutdown as required by the reactor protection system.

LDCN-FSAR-99-053 (SE 99-0039)

This LDCN provided for resolution of Thermo-Lag and the Thermo-Lag Reduction Plan. Specifically, the FSAR was revised to more clearly define in what fire areas mineral insulated cable is credited as a fire barrier, and to reference Fire Protection Engineering Evaluation FPF 1.11, Item 2, as the basis for fire protection qualification of Whittaker mineral insulated cable.

Safety Evaluation Summary

It was concluded from the safety evaluation that previous Thermo-Lag design changes and their associated safety evaluations have already justified that the mineral insulated cable is a qualified electrical cable for its plant location and equipment support functions. Thus, this safety evaluation only covers the fire rating aspect.

In summary, the change uses mineral insulated fire-rated cable as a new type of raceway fire barrier. The mineral insulated cable provides the fire resistance mandated by the NRC and helps to ensure PFSS. These changes are in accordance with fire protection License Condition 2.c (14) and do not constitute an unreviewed safety question.

LDCN-FSAR-99-064 (SE 99-0044)

This LDCN provided for a change to FSAR section 12.5.3.3, "Radiological Survey Procedures," to delete the specified weekly and annual routine surveys.

Safety Evaluation Summary

It was concluded from the safety evaluation that the weekly and annual routine surveys currently discussed in the FSAR are not required by any other licensing basis document. 10 CFR 20 only requires that the licensee perform surveys that are reasonable under the circumstances to evaluate radiation and contamination levels and the potential radiological hazards. In addition, the weekly and annual routine surveys are not derived from nor related to any design basis accident evaluation.

LDCN-FSAR-99-064 (SE 99-0045)

This LDCN provided for a revision to FSAR section 12.5.3.7, "Personnel Dosimetry Procedures." The occupational radiation exposure monitoring requirements for individuals in the controlled area and restricted areas were revised to reflect a recent NRC policy change. The policy change raises the monitoring criteria for minors from 50 to 100 millirem in a year and for declared pregnant women from 50 to 100 millirem during their pregnancies.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed change, which simply reflects the recent change to the 10 CFR 20 monitoring criteria, does not constitute an unreviewed safety question and conforms to the requirements of the WNP-2 Operating License.

LDCN-FSAR-99-080 (SE 99-0053)

This LDCN provided for a change to the FSAR to correct the factors of safety for the overturning and sliding of the standby service water pumphouses during a safe shutdown earthquake. The existing FSAR values did not match values in the recent corresponding design calculations.

Safety Evaluation Summary

It was concluded from the safety evaluation that the corrected factors of safety are lower than those currently in the FSAR. However, the corrected factors are greater than those specified in the NRC Standard Review Plan. Therefore, there is adequate margin of safety to ensure that the standby service water pump house does not slide or overturn in a design basis earthquake. In addition, the service water system and the ultimate heat sink continue to perform their design basis functions. All other seismic design parameters are unchanged.

LDCN-FSAR-99-081 (SE 99-0054)

This LDCN provided for the following changes to the FSAR:

- Section F.2.2.3, "Fireproof Coatings," was revised to add a discussion of crediting TS-5269 and reference FPF 1.12, Item 2, for overall qualification of Thermo-Lag.
- Section F.4.4.4, "Fire Area Analyses," was revised to add a discussion of Thermo-Lag on support TS-5269 as a credited feature.
- Section F.7.3, "Calculations/Technical Memos," was revised to add a reference to CE-02-89-20.
- Section F.7.6, "Fire Protection Engineering Evaluations," was revised to add FPF 1.12, Item 2, as a new reference.

Safety Evaluation Summary

It was concluded from the safety evaluation that no unreviewed safety questions were identified as a result of any of these changes. Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers," and NUMARC/NEI fire testing have determined that Thermo-Lag may not provide the required hourly rating necessary for raceway fire barriers. Therefore, the plant modification series 92-0276 result in the elimination of WNP-2's reliance on Thermo-Lag as a PFSS raceway fire barrier. However, PFP 1.12, Item 2, justifies why Thermo-Lag can safely remain credited as steel fireproofing on certain reactor building Division 2 instrument tube supports and cable tray support TS-5269. The thickness and design of the Thermo-Lag is adequate to ensure steel warpage does not occur under design basis fire conditions.

To ensure operability is maintained, PPM 15.4.9, "Fire-rated Thermo-Lag Structural Steel Fireproofing Operability Inspection," is being issued which periodically surveils the Thermo-Lag coated instrument tube supports and TS-5269. These are existing plant features and do not require new Thermo-Lag installations. Rework of damaged and/or degraded Thermo-Lag installations will be performed as specified in existing approved plant procedure PPM 10.25.89, "Thermo-Lag Fire Barrier Installation and Inspection."

In summary, the primary focus of this change is to justify that the existing Thermo-Lag design over instrument tube supports and tray support TS-5269 ensures PFSS capability is maintained. These changes are in accordance with fire protection License Condition 2.c (14), do not affect the operation of any safety-related or safe shutdown system, do not affect previously analyzed FSAR Chapter 3, 6, or 15 accidents, transients, or equipment malfunctions important to safety and, therefore, do not result in an unreviewed safety question.

LDCN-COLR-98-135 (98-106)

This LDCN provided for Revision 1 to the Core Operating Limits Report (COLR) 98-14 in order to change certain minimum critical power ratio (MCPR) values. The changes were made necessary by the discovery of errors in the vendor's transient analysis for Cycle 14.

Safety Evaluation Summary

It was concluded from the safety evaluation that the modifications to certain MCPR limits were developed with approved NRC methodology. The assumptions used in the supporting analysis are those identified in the relevant safety evaluation reports. This change does not increase the probability of accidents previously evaluated for WNP-2 since the core limits defined in the modified Cycle 14 COLR were observed. The change does not increase the consequences of equipment malfunctions previously evaluated in the safety analyses report because the mechanical, thermal-hydraulic, and LOCA design criteria imposed on the fuel and the core maintain the level of protection previously achieved during accidents. In addition, the fuel cladding and coolable geometry of the fuel is not challenged by the change. Continued operation of Cycle 14 within the thermal limits specified in COLR 98-14, Revision 1 does not reduce the margin of safety as defined in the bases for the core design related Technical Specifications.

LDCN-EP-99-006 (SE 99-0031)

This LDCN provided for Revision 23 to the Emergency Plan to provide updated descriptions of the separation of the Benton/Franklin County Emergency Operations Center (EOC). Franklin County opted to operate an independent EOC and no longer participate in a combined EOC with Benton County. Plant and County notification procedures were changed to reflect the revised notification process. Additional revisions to the Emergency Plan include changing the Technical Support Center (TSC) Radiation Data Coordinator position title to TSC

Chemistry/Effluent Manager, and adding responsibility for providing chemistry advice to the TSC Manager. In addition, the description of fire brigade turnout stations and first aid locations was updated and first aid kits are being referred to as first responder kits.

Safety Evaluation Summary

It was concluded from the safety evaluation that an unreviewed safety question is not created as a result of this change. The proposed activity does not increase the probability of occurrence of an accident evaluated previously in the safety analysis report because the actions implemented by the proposed changes are performed after an accident occurs, such as notification of offsite agencies or recommended protective actions to offsite agencies. In addition, the margin of safety as defined in the basis for any Technical Specification is not reduced as a result of implementing these changes as neither the Emergency Plan or the notification scheme are addressed in the basis for the Technical Specifications.

LDCN-EP-99-075 (SE 99-0047)

This LDCN provided for Revision 24 to the Emergency Plan to change the Emergency Action Levels (EALs) contained in Table 4-1, "Emergency Classification Initiating Conditions," and PPMs 13.1.1, "Classifying the Emergency," and 13.1.1A, "Classifying the Emergency – Technical Bases." These changes remove specific reference to procedure numbers and instead provide a word description of the plant condition that constitutes the requirement to declare the specific EAL met. The reference to procedures in EALs is not needed since the method of operation is to enter the appropriate procedure, and then, when the EAL condition is met, declare the appropriate emergency classification. Referencing the procedure number in the EAL does not help in dealing with the emergency since the procedure has already been entered prior to declaring that the EAL condition has been met. Therefore, references to procedures in the EALs were deleted.

Changes are also being made to sections 3.1, "Coordination of Support Organizations," 5.3.2, "Emergency Dose Projection System," and 8.3, "Independent Audit of the Emergency Preparedness Program," of the Emergency Plan to provide additional clarification of responsibilities.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes to the EALs provide a word description, vice procedural reference, for plant conditions. These changes make it easier for personnel to identify the appropriate EAL for a given plant condition. The EALs are used to determine the appropriate emergency classification level for the plant. From the classification level, actions such as evacuation, sheltering, etc., are determined. Since none of these actions are credited for accident analyses and these actions are not affected by the changes; an unreviewed safety question does not exist.

LDCN-ODCM-99-023 (SE 99-0015)

This LDCN provided for a change to the Offsite Dose Calculation Manual (ODCM) to provide clear direction for primary containment purging and venting through standby gas treatment using the 24-inch, 30-inch, or two-inch exhaust valves. In addition, definitions for venting and purging that were previously included in the Technical Specifications were added. Clarification was also added to Note h in Table 6.2.2.1.2-1, "Radioactive Gaseous Waste Sampling and Analysis Program," relative to the type of vent to which it applies. References to the drywell were deleted, and where appropriate, were replaced with containment since this requirement for operability has always been interpreted to apply to both the drywell and wetwell purges.

Safety Evaluation Summary

It was concluded from the safety evaluation that the proposed changes to the ODCM result in a clear definition of the requirements for standby gas treatment when purging or venting the primary containment. This activity was part of the original Technical Specifications and licensing basis. Sufficient requirements exist to ensure that the bases requirement to maintain releases to within 10 CFR 20 requirements is met. As such, the proposed activities do not result in a change to the Technical Specifications or the creation of an unreviewed safety question.

LDCN-TSB-99-010 (SE 99-0007)

This LDCN provided for a change to the Technical Specifications Bases Section B 3.7.1, "Standby Service Water (SW) System and Ultimate Heat Sink (UHS)." The clarification provides consistency in applying LCO 3.0.6 to conditions that result in loss of room coolers for equipment required to safely shutdown the plant following a design basis accident or transient. The specific changes are as follows:

- The Bases Background section was revised to match FSAR Section 9.2.7.1, "Design Bases."
- The addition of a statement to the Bases Background section that heat exchangers and room coolers which have service water flow through them post accident perform a support function to the service water design function.
- The Bases LCO section was revised to clarify the affect of isolating service water to component/systems on service water operability.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes add clarifying language to the Technical Specifications Bases to ensure consistent implementation of LCO 3.0.6 when the loss of a room cooler function occurs. These changes do not alter the intent of the Technical Specification LCO requirements nor change the intent of the use of LCO 3.0.6.

This change includes addressing critical electrical switchgear room coolers, electrical equipment room coolers, and other heat exchangers similar to the ECCS room coolers. The design function of the service water system is to remove heat from plant systems that are required for a safe reactor shutdown following a design basis accident or transient. The room coolers/heat exchangers for the areas that contain these plant systems support the safety function of the service water system.

Technical Specification LCO 3.7.1 addresses required actions associated with complete loss of one service water subsystem and allows 72 hours to restore the service water subsystem or be in Mode 3 within 12 hours. LCO 3.0.6 allows that when a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with the supported system are not required to be entered. This change does not reduce the margin of safety as defined in the basis for any Technical Specifications. The Technical Specifications currently allow completion of the service water LCO required actions under the provisions of LCO 3.0.6 before the required actions of the above LCOs are required to be taken. This envelopes the loss of single or multiple room coolers in the same division supplied by the respective service water subsystem.

LDCN-TSB-99-077 (SE 99-0052)

This LDCN provided for a change to Technical Specifications Bases Section 3.10.2, "Reactor Mode Switch Interlock Testing" to include the allowance of maintaining the mode switch in a position other than required by Technical Specification Table 1.1-1, "Modes." This allows support of ongoing or planned testing if the required testing is scheduled to start within the next 12 hours. During the time period the mode switch is not in the Table 1.1-1 position, Surveillance Requirements 3.10.2.1 and 3.10.2.2 shall continue to be met.

Safety Evaluation Summary

It was concluded from the safety evaluation that maintaining the mode switch in other than the Technical Specification Table 1.1-1 position for limited periods of time between concurrent mode switch interlock tests does not represent an unreviewed safety question. This is based on the fact that having the mode switch in other than the Technical Specification Table 1.1-1 position has already been reviewed by the NRC and found to be acceptable for a non-specified period provided that the defeated interlocks are administratively controlled via performance of surveillances at 12-hour and 24-hour frequencies. The change to the Technical Specifications Bases maintains these requirements while allowing the mode switch to be maintained in the position required to support testing scheduled to commence within the next 12 hours.

2.6.3 Miscellaneous Changes

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

Calculation Modification Record CMR 99-0021 (SE 99-0004)

This calculation modification record provided for the modification of the core monitoring system (Powerplex) input deck to include a 0.03 penalty to compensate for potential non-conservative monitored minimum critical power ratios (MCPR) values for SVEA-96 reload fuel in WNP-2 at certain highly top peaked axial power distributions.

Safety Evaluation Summary

It was concluded from the safety evaluation that this change does not result in an unreviewed safety question. The safety significance associated with this change is that application of the XL-S96 critical power ratio (CPR) correlation for monitoring CPRs in SVEA-96 reload fuel for top-peaked axial power shapes can result in non-conservative monitored CPR values. Evaluations of this defect have been performed by Asea Brown Boveri (ABB) CENO. It was determined that the use of the correlation for determining MCPR values for Cycles 12, 13, and 14 was acceptable. The use of this correlation for monitoring CPR was also found acceptable for Cycles 12 and 13. For cycle 14, the use of this correlation for monitoring CPR was found potentially unacceptable for exposures beyond 5600 MWD/MTU and specifically for highly top peaked axial power shapes. The evaluation determined that a decrease in MCPR as determined by the core monitoring system should be implemented for cycle 14 exposure beyond 5600 MWD/MTU.

Configuration Document Change Request CDCR 98-02-007 (SE 98-026)

This configuration document change request provided for drawing changes to reflect that the as-built rating of grounding transformer HPCS-TR-GND is 10 KVA. Since Diagram E5202-2 is also included in the FSAR, changes to it are considered "de facto" design changes to the facility, as described in the safety analysis report, and are evaluated pursuant to 10 CFR 50.59.

Safety Evaluation Summary

It was concluded from the safety evaluation that HPCS-TR-GND provides input power to ground fault alarm relay HPCS-RLY-59N. Only a failure within the HPCS distribution system can result in a ground fault alarm, and HPCS system failures are compensated through operation of redundant systems. Passive failure or degradation of the grounding transformer results in the loss of the ground alarm function and loss, or degradation, of the generator neutral. This failure does not affect the operation of the generator until a ground fault occurs elsewhere in the distribution system, the consequences of which would be no worse than a complete loss of HPCS, which is an analyzed single failure condition. Therefore, changing the size of the HPCS generator grounding transformer from 7-½ to 10 KVA cannot affect the capability of the HPCS diesel-generator to perform its safety function. Additionally, the calculated minimum KVA rating for continuous operation of the grounding transformer is 6.864 KVA. The Westinghouse Electrical Transmission and Distribution Reference Book states in part that, "In general, it is preferable to be conservative on the transformer rating in order that its reactance is not an appreciable factor." The presently depicted 7-½ KVA rated transformer is capable of continuously carrying the calculated ground fault current with margin. The installed 10 KVA rated transformer is capable of sustaining the same ground fault current with even greater margin.

Configuration Document Change Request CDCR 98-05-020 (SE 98-094)

This configuration document change request provided for a drawing revision to reflect the addition of a cap on the end of the drain line downstream from SW-V-194. In addition, this request provided for a change to the description of SW-V-23A from a globe valve to a gate valve to reflect actual field conditions.

Safety Evaluation Summary

It was concluded from the safety evaluation that SW-V-194 provides an ASME pressure boundary for the SW supply to DCW-HX-1A1. This valve is leaking slightly and the cap has been added to keep the water from filling the EDR sump with SW. The leakage has been evaluated and does not adversely impact the ability of SW to meet its safety function. This cap is not relied on to provide a pressure boundary but is for housekeeping purposes only.

Valve SW-V-23A provides outlet isolation and throttling for RHR-HX-2A. Using a gate valve for this application does not impact its ability to perform its safety function. Most of the throttle valves on the SW system are gate valves. This valve sufficiently ensures adequate flow is maintained to the heat exchanger. Flow balance procedures for SW ensures that sufficient flow is provided to all SW loads to ensure equipment receives adequate cooling.

Configuration Document Change Request CDCR 99-03-013 (SE 99-0040)

This configuration document change request provided for the replacement of backflow preventer potable water valve PWC-V-162 and the required drawing change to accurately reflect as-built conditions and the removal of equipment piece number PWC-V-172.

Safety Evaluation Summary

It was concluded from the safety evaluation that analysis and field testing of the proposed backflow preventer has shown there is no increased risk to safe plant operation due to this activity. All plant systems and components required to mitigate the consequences of accidents previously evaluated are not affected by this installation. Therefore, the activity does not increase the probability of occurrence or the consequences of an accident evaluated previously in the licensing basis documents.

Configuration Document Change Request CDCR 99-10-008 (SE 99-0059)

This configuration document change request provided for a drawing revision to show SLC-V-16 maintained in a CLOSED position and SLC-V-17 in a LOCKED CLOSED position. The valves, their operation, orientation, and physical characteristics are not changed. Neither valve is a PCIV.

Safety Evaluation Summary

It was concluded from the safety evaluation that the change in the position designation described for these valves does not pose an unreviewed safety question. A review of the standby liquid control system design and operation determined that this change does not impact the function or operation of this system. Review of the FSAR analysis for those accidents that the standby liquid control system is credited with mitigating determined that this change has no adverse impact on the accident probability or severity. In addition, this change does not constitute a change to the Technical Specifications or Operating License.

Plant Procedure PPM 6.2.5 (SE 99-0030)
Plant Procedure PPM 1.3.40 (SE 99-0030)

These procedures are refueling activity procedures and describe the process of handling, preparing, inspecting and installing new reactor fuel channels on the refuel floor. It was necessary to revise these procedures to allow movement of non-irradiated loads over irradiated fuel with secondary containment and standby gas treatment (SGT) unavailable. Technical Specifications do not require secondary containment or SGT to be operable when moving non-irradiated fuel in the fuel pool.

Safety Evaluation Summary

It was concluded from the safety evaluation that by requiring a mandatory 90 day cooling/decay time for spent fuel before moving loads above irradiated fuel assemblies, the safety functions of SGT, control room emergency filtration, and secondary containment are met. This activity is not governed by the Technical Specifications; therefore, there is no requirement to meet single failure criteria. In addition, these systems are not being relied upon to mitigate an accident.

Plant Procedure PPM OSP-RPV-R801 (SE 98-097)

This procedure provided for instructions for performing a pressure test on the ASME Code Class 1 piping and components to satisfy the ASME Section XI Inservice Inspection and leakage pressure testing requirements. The procedure was revised to disable the main steam isolation valve (MSIV) scram interlock and the reactor pressure vessel (RPV) high-pressure scram during execution of the RPV leak test. The MSIV scram interlock provides for a reactor scram with the mode switch not in run, two or more MSIVs not full open, and RPV pressure greater than or equal to 1053 psig. The RPV high-pressure scram initiates a reactor scram at 1060 psig. During the RPV leak test, all rods are inserted and the RPV is filled with coolant. The associated pressure switches sense the additional water head present and when added to the RPV test pressure can cause a scram signal. Historically, the RPV leak test has been performed with a scram inserted. This prevented a scram from occurring during initial RPV leak tests. Since it is now desired to perform CRD scram time testing concurrent with the leak test, a scram cannot be inserted during the test. Disabling the high-pressure scram function minimizes the possibility of a scram occurring as a consequence of performing the RPV leak test.

Safety Evaluation Summary

It was concluded from the safety evaluation that the changes made to the procedure do not constitute an unreviewed safety question. Disabling the high-pressure scram during the RPV leak test does not impact any important to safety equipment required during this special Mode 4 condition. Special containment LCOs in accordance with Technical Specification 3.10.1, "Special Operations" is not affected. The low-pressure cooling systems used in the event of a large leak are also not affected. In addition, core reactivity is not impacted by disabling this scram function since all rods are in (with the exception of one as permitted by the one rod out interlock during concurrent scram time testing) as they would be during any other Mode 4 activity.

Work Order NZGO (SE 99-0029)

This work order provided for: 1) temporary connection of chemical cleaning equipment to the stator coil cooling water (SCW) system; 2) the cleaning of the SCW system including the main generator water cooled stator bars by proprietary chemical process; and 3) the removal of the chemical cleaning equipment in Modes 1, 2, 3, 4, and 5. While the chemical cleaning can be done with the generator on-line or off-line (any mode), the preferred method is to clean while the generator is at full power in order to measure the efficiency of the cleaning process. Chemical addition will be accomplished through the use of vendor supplied injection pumps that are connected to the system in accordance with PPM 1.3.9, "Temporary Modifications." The chemical cleaning process (patented by ABB) uses an aqueous solution of disodium EDTA, hydrogen peroxide, Cuproplex Inhibitor I505, Activator A505, and sodium sulfate. The SCW system demineralizers continuously remove the dissolved copper oxide and chemicals in solution. Depleted demineralizers were changed out with new resins.

Safety Evaluation Summary

The SCW system, main generator, and all associated chemical cleaning equipment remain non-safety-related, Quality Class 2 or G, and Seismic Class 2. The proposed activity does not challenge the design of SCW system. The proposed activity does not impact any existing design basis accident, PSA, or IPE analysis evaluation.

The consequences of any accident that could occur during the cleaning process are not compounded by the presence or use of the chemical cleaning agents in the SCW system. The SCW system is not cross-connected to any safety-related system. The system is manually isolated from the demineralized water system and hydrogen system during the cleaning process. In addition, any SCW system leakage would go to the floor drain processing (FDR) system where chemicals would be removed by the FDR demineralizers prior to recycle for CST use. The chemicals added to the SCW system were administratively controlled in accordance with plant procedures and did not come in contact with the reactor coolant.

Work Order PTS6-16 (SE 99-0051)

This work order provided for an alternate safety-related power supply (temporary power) to fuel pool cooling pump FPC-P-1A (FPC-M-P/1A) so that it is available during the E-SM-7 bus outage. This was accomplished by replacing the normal source (E-MC-7BB/9B) with an alternate source taken from (E-MC-8BB/1CL (spare). This change ensures FPC-P-1A (FPC-M-P/1A) remains operational while spent fuel pool heat load is high. The crosstie of safety-related divisions is permitted by Technical Specification Section 3.8.8, "Distribution Systems – Shutdown" in Modes 4 and 5.

Safety Evaluation Summary

It was concluded from the safety evaluation that the temporary modification of the power source to PPC-P-1A (FPC-M-P/1A) does not affect the ability of the fuel pool cooling system or any other system to perform its intended safety function. No tests or experiments are involved and no changes are made to the Technical Specifications. Therefore, an unreviewed safety question does not exist.

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 8.3.411 (SE 99-0012)

This procedure provides instructions for performing the post-modification testing of the OPRM installation.

<u>Safety Evaluation Summary</u>

It was concluded from the safety evaluation that no unreviewed safety questions exist as a result of the procedure for post-modification testing of the OPRM. The OPRM system works in conjunction with the LPRM and the APRM systems to detect and mitigate the power oscillations. The OPRM interfaces with the electronic portion of these monitoring systems. The testing of this procedure provides assurance that the interfaces and connections to the electronic portion of the systems are within the design capabilities of the equipment, and therefore, within the previously evaluated accident mitigating functions. This ensures that the accident mitigating functions of the APRM are not affected by the operation of the OPRM and provides assurance that the consequences of an accident are not increased as a result of OPRM installation. In addition, the only protective function of the OPRM system is to initiate a scram which is not required under the test conditions. Since the reactor was shutdown for this testing, the protective feature was not needed.

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-159531-00 (Area of Increased Awareness (AIA) Entry Plan)

The original commitment description reads, "The Supply System will continue this approach, by requiring that a startup plan (AIA Entry Plan) developed in accordance with a POC approved procedure be in place prior to exceeding 25% thermal power."

The commitment was made in LER 92-037-03 [Reference: Letter GO2-94-179, dated July 28, 1994, JV Parrish (Supply System) to NRC, Licensee Event Report 92-037-03].

This commitment was revised. Revision 26 of the Operational Quality Assurance Program Description eliminated the requirement for POC to review plant procedures. Eliminating POC review of the AIA Entry Plan procedure does not affect the commitment to have an AIA Entry Plan. An AIA Entry Plan is still required in an approved plant procedure. Therefore, the commitment is revised to read, "The Supply System (Energy Northwest) will continue this approach, by requiring that an AIA Entry Plan be developed in accordance with an approved procedure and be in place prior to exceeding 25% thermal power."

RCC-104285-00 (Rosemount Transmitters)

The original commitment description included a commitment to an enhanced monitoring program for Rosemount dp Transmitters. A subsequent letter included a commitment to continue the enhanced monitoring program indefinitely.

The commitment was made in response to IEB 90-01 [References: Letter GO2-93-055, dated March 8, 1993, GC Sorenson (Supply System) to NRC, Response to IEB 90-01, Supplement 1; and, Letter GO2-94-124, dated May 23, 1994, JV Parrish (Supply System) to NRC, Response to Request for Additional Information].

This commitment was deleted. The enhanced transmitter monitoring program was established to observe transmitter calibration data for transmitters in service above 500 psi that have not exceeded a vendor recommended psi/month threshold. Vendor research indicated that after a length of time at pressure, the oil loss problem would not occur as the transmitter was aged in service. Transmitters in the WNP-2 enhanced monitoring program reached the psi/month threshold in late 1995. Training has been provided to I&C

technicians for oil loss problems with transmitters. This training is part of ongoing maintenance qualifications. Considering that transmitters have exceeded the vendor recommended monitoring time, and I&C technicians are qualified to observe transmitter problems, the enhanced monitoring program is no longer necessary.

RCC-10824-00 (QA Surveillance Deficiencies)

The original commitment description reads "Plant QA Procedure PQA-03, Conduct of QA Surveillances, will be revised to provide direction to evaluate surveillance findings against plant problem reporting requirements."

The commitment was made in LER 88-028 [Reference: LER 88-028, dated September 22, 1988, CM Powers (Supply System) to NRC, Licensee Event Report 88-028].

This commitment was deleted. At the time the commitment was made, QA surveillance deficiencies were recorded on Quality Finding Reports, which had no tie to the corrective action process and were not reviewed for reportability. Currently, QA surveillance findings are delineated on Problem Evaluation Requests as part of the site-wide corrective action program. The corrective action program requires that all Problem Evaluation Requests are reviewed for reportability. A separate commitment for QA surveillance deficiencies is not necessary.

RCC-116677-01 (Additional Exposure Controls)

The original commitment description reads, "Revise PPM 11.2.2.5 to require additional exposure controls for non-routine high rad area entries greater than 50 mrem per task."

This commitment had been previously revised to read [Reference: RCC-116677-00], "Implement the controls of PPM 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 original commitment was made in response to NRC Inspection Report and NOV 95-16 [Reference: Letter GO2-95-137, dated July 21, 1995, JV Parrish (Supply System) to NRC, Response to Apparent Violations].

This commitment was deleted. The intent of this commitment was to provide an additional barrier to unnecessary exposures and potential overexposures by identifying an individual responsible for tracking job status against anticipated exposure (i.e., the PIC, or "Person in Charge"). The aim was to ensure active participation of work group supervision in management of the work and in dose control. The radiation exposure control climate has changed dramatically in the last few years to the extent that Additional Exposure Controls (AEC) are no longer necessary because:

- The transition in work management to work teams which include RP personnel has led to a greater degree of diligence in ALARA planning, dose tracking, and dose control;
- The increased emphasis by management toward dose budgets and daily doses has led to site-wide improvements in the ownership and personal awareness of individual and work group doses, and to the evolution of a questioning attitude toward unnecessary doses;
- AEC should not be confined to a single regulatory commitment since there are alternative and more efficient means to ensure AEC, such as greater attentiveness and intelligence in work planning, ALARA planning, Radiation Work Permits, and radiological oversight and control; and,
- Technological improvements and advances in personnel dosimetry and dose tracking, such as wireless remote monitoring, has provided, and will increasingly provide, a better ability to more closely track and control individual exposures.

RCC-144467-00 (Work in Radiological Areas)

The original commitment reads, "Require workers to notify Health Physics personnel prior to starting work in radiological areas."

The original commitment was made in response to NRC Inspection Report 97-16 and NOV [Reference: Letter GO2-97-210, dated November 20, 1997, PR Bemis (Supply System) to NRC, NRC Inspection Report 97-16, Response to Notice of Violation].

This commitment was revised. Instead of requiring all workers to notify Health Physics prior to starting work, the commitment was revised to read, "Prior to entering the Radiological Controlled Area, individuals should have a face-to-face meeting with a Health Physics staff member if any of the following conditions exist:

- The Radiation Work Permit requires a pre-job briefing;
- The work location is a high, high-high or very high radiation area;
- The work location is in a contaminated area; or,
- The work activity has the potential to change the radiological condition in the work location (e.g., activities such as breaching a contaminated system, welding or grinding on contaminated components, movement or transfer of radioactive material, draining or venting radioactive systems).

Appendix A

Annual Personnel Radiation Exposure
Work and Job Function Report
Calendar Year 1999

WNP-2 ANNUAL OPERATING REPORT
WORK AND JOB FUNCTION REPORT
CALENDAR YEAR 1999

This report was produced with direct reading dosimeter data.

Number of Persons Receiving Over 100 millirem is 463.

Total man-rem is 144.681

		Number of individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contractors and Others
OPERATIONS & SURVEILLANCE	Maintenance Personnel	52.25	2.85	23.55	8.633	0.337	2.338
	Operating Personnel	27.92	0.00	0.99	7.705	0.000	0.158
	Health Physics Personnel	9.37	0.28	0.19	1.559	0.033	0.082
	Supervisory Personnel	3.75	0.00	0.53	0.401	0.000	0.080
	Engineering Personnel	6.77	1.45	4.02	0.601	0.084	0.328
ROUTINE MAINTENANCE	Maintenance Personnel	51.96	1.59	102.67	25.914	0.981	36.858
	Operating Personnel	0.59	0.00	0.10	1.202	0.000	0.105
	Health Physics Personnel	30.92	0.83	12.11	8.877	0.129	3.400
	Supervisory Personnel	1.07	0.00	0.41	0.500	0.000	0.159
	Engineering Personnel	5.25	1.39	7.70	1.382	0.248	1.864
INSERVICE INSPECTION	Maintenance Personnel	1.53	0.00	9.46	0.845	0.000	3.346
	Operating Personnel	1.14	0.00	0.00	2.086	0.000	0.000
	Health Physics Personnel	0.41	0.00	0.39	0.330	0.000	0.319
	Supervisory Personnel	0.08	0.00	0.13	0.043	0.000	0.099
	Engineering Personnel	1.90	1.20	2.54	0.608	0.318	0.399
SPECIAL MAINTENANCE	Maintenance Personnel	5.25	0.12	49.44	2.173	0.082	9.512
	Operating Personnel	0.03	0.00	0.00	0.107	0.000	0.000
	Health Physics Personnel	0.55	0.00	0.13	0.783	0.000	0.066
	Supervisory Personnel	0.06	0.00	0.93	0.017	0.000	0.177
	Engineering Personnel	0.46	0.11	2.77	0.112	0.019	0.388
WASTE PROCESSING	Maintenance Personnel	0.30	0.41	0.00	0.164	0.309	0.000
	Operating Personnel	0.00	0.00	0.88	0.000	0.000	0.182
	Health Physics Personnel	0.13	0.00	0.00	0.186	0.000	0.000
	Supervisory Personnel	0.01	0.00	0.00	0.006	0.000	0.000
	Engineering Personnel	0.00	0.00	0.00	0.000	0.000	0.000
REFUELING	Maintenance Personnel	16.30	0.76	2.85	13.024	0.155	1.760
	Operating Personnel	0.48	0.00	0.02	0.438	0.000	0.018
	Health Physics Personnel	1.87	0.89	2.08	0.704	0.276	0.414
	Supervisory Personnel	3.19	0.00	0.00	0.786	0.000	0.000
	Engineering Personnel	1.43	0.97	0.86	0.118	0.091	0.028

WNP-2 ANNUAL OPERATING REPORT
WORK AND JOB FUNCTION REPORT
CALENDAR YEAR 1999

		Number of individuals			Year to Date Dose		
		Station Employees	Utility Employees	Contractors and Others	Station Employees	Utility Employees	Contractors and Others
TOTAL	Maintenance Personnel	127.59	5.73	187.97	50.753	1.864	53.814
	Operating Personnel	30.16	0.00	1.99	11.538	0.000	0.463
	Health Physics Personnel	43.25	2.00	14.90	12.439	0.438	4.281
	Supervisory Personnel	8.16	0.00	2.00	1.753	0.000	0.515
	Engineering Personnel	15.81	5.12	17.89	2.821	0.760	3.007
GRAND TOTAL		224.97	12.85	224.75	79.304	3.062	62.080

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

- Installation of penetration seals
- Painting and labeling of the 548 elevation reactor building
- Darmatt installation
- Residual heat removal pump 2B modification
- Residual heat removal valve cable termination
- Crossunder piping thermal spray coating
- Thermo-Lag removal
- Reactor core isolation cooling valve baseline post-modification testing