ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket: 50-397
License: NPF-21

Report: 50-397/9603

Licensee: Washington Public Power Supply System

Facility: Washington Nuclear Project-2

Location: 3000 George Washington Way
P.O. Box 968, MD 1023
Richland, Washington

Dates: February 18 - March 30, 1996

Inspectors: R. C. Barr, Senior Resident Inspector
G. D. Replogle, Resident Inspector
D. E. Corporandy, Project Inspector

Approved by: H. J. Wong, Chief, Branch E
Division of Reactor Projects

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/96-03

This routine announced inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Compared to previous reactor shutdowns, the licensee's conduct of the reactor shutdown this inspection period was generally better coordinated. Three-way communications and second person verifications were considered improved when compared to past performance. However, several weaknesses were identified associated with procedural adherence and usage, resulting in a cited and a noncited violation. These examples included: (1) the failure to verify the shutdown procedure prior to use (self-identified, noncited violation); (2) the failure to perform a required chemistry sample of the reactor coolant (NRC-identified, cited violation); and (3) the failure to perform the shutdown procedure in sequence or provide specific management direction to the contrary (Section O1.2).

- The inspectors identified several instances where operations did not exhibit a strong questioning attitude when unexpected alarms or unwarranted operating conditions occurred. The conditions included a main generator over-excitation alarm, a high turbine exhaust temperature alarm, and an entry into emergency operating procedures as a result of lowering reactor vessel water level from 36 to 25 inches prior to the reactor scram. A combination of confusing procedural guidance, inadequate operator knowledge, and out-of-tolerance equipment contributed to some of the alarm conditions (Section O1.2).

Maintenance

- Maintenance tasks and surveillance testing were generally performed and documented properly (Section M1.1).

Engineering

- In three instances the clearance order process was inappropriately used to permanently remove equipment from service, instead of completing formal modifications to the facility. This resulted in the Final Safety Analysis Report (FSAR) not being revised in three instances. Moreover, in the last 12 weeks NRC inspectors have identified five instances of not appropriately updating the FSAR, which may suggest that the system engineers' knowledge of their assigned system(s), the FSAR, what

constitutes a change to the facility, and current understanding of system design bases and deviations is not adequate (Section E1.1).

- The inspectors identified a violation with two examples in which conditions adverse to quality were not promptly documented and resolved (Section E8.1).

- Assigned system engineers were not aware that operating procedures allowed extended periods of low speed reactor feedwater turbine (RFT) operation which could adversely affect turbine performance (Section O1.2).

Plant Support

- Dose reduction efforts, such as system flushes and shielding installation, continue to be effectively implemented, resulting in reduced month-to-month radiation exposures (Section R1.1).

Report Details

Summary of Plant Status

Due to excess electrical generation capacity in the Northwest, the plant operated at approximately 60 percent power from February 18-26 and returned to 100 percent from February 26-29. On February 29 operators reduced reactor power to 60 percent in preparation for reactor shutdown. On March 1, from approximately 60 percent power, reactor shutdown was started and completed on March 2 with entry into Mode 3, completing 243 days of continuous operation and the first continuous operation from refueling to refueling. The plant entered Mode 4 on March 3, where the plant remained during the remainder of the inspection period.

I. Operations

01 Conduct of Operations

01.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. The conduct of operations was generally professional and safety-conscious. Communications and second person verifications were improved. During a reactor shutdown, the inspectors observed weaknesses with procedural adherence and procedure adequacy, the quality of control room logs, operator knowledge of alarm setpoints, and the oversight of oncoming reactor operators (ROs). These observations are detailed in the following sections.

01.2 Reactor Shutdown Observations

a. Inspection Scope (71707)

The inspectors observed selected portions of shutdown activities from the control room on March 1-2. The inspectors observed parts of the evolution in the control room and reviewed operator logs related to the remainder of the shutdown process.

b. Observations and Findings

Communications: The inspectors observed that the control room staff consistently demonstrated good three-way communication skills during the shutdown and continued to show improvement in other communications.

Second Person Verifications: At one point during the rod insertion sequence, a trainee inadvertently selected the wrong control rod for insertion. This error was immediately corrected by the shift technical advisor (STA) performing the second person verification. The onshift

reactor operator was nearby monitoring the trainee, but did not have time to correct the trainee before the STA identified the selection error. The trainee then selected and moved the correct control rod. The control room supervisor (CRS) subsequently provided additional guidance to the trainee to enhance his skills. The inspector considered that the STA had performed the second person verification duties in an excellent manner.

Procedural Adherence and Usage: The inspectors observed that operators conducted the plant shutdown in accordance with Plant Procedure Manual (PPM) 3.2.1, "Normal Shutdown to Cold Shutdown." Relatively early during the shutdown, the onshift STA identified that the copy of PPM 3.2.1 which was being used to conduct the shutdown did not contain the most recent procedure change. A temporary change notice had been issued to PPM 3.2.1 on February 27, 1996. The temporary change notice addressed the cleaning of condensate demineralizers. Procedures for plant shutdown had been put together prior to the issuance of the change and the licensee had failed to verify and document by signature, a licensee procedural requirement, that the appropriate version of the procedure was being used. Due to early identification during the shutdown, this error had no effect on the evolution. As corrective action, the licensee implemented a performance improvement plan for the individual that failed to verify the document. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-397/9603-01).

During the review of control room logs on March 4, the inspectors identified that the licensee failed to perform a chemistry sample required by Technical Specifications (TS) Surveillance Requirement 4.4.5 and implemented by licensee procedure PPM 3.2.1. TS Surveillance Requirement 4.4.5 required demonstration that the primary (reactor) coolant was within specified limits by performance of a sampling and analysis program specified in Table 4.4.5-1. Table 4.4.5-1, Item 4.b, specifies sampling the reactor coolant between 2 and 6 hours after a power change of greater than 15 percent in 1 hour, as required by Action c. Action c requires the performance of the sampling and analysis requirements of Table 4.4.5-1, Item 4.b, when in operational Condition 1 or 2 and with a power level change of greater than 15 percent of rated thermal power in 1 hour.

In operational Condition 1 and as a result of reducing the speed of the recirculation pumps to 15 hertz at 11:30 p.m. on March 1, reactor power was changed by approximately 17 percent; however, a primary coolant sample was not taken nor isotopic analyses for iodine performed within 2-6 hours. The failure to take and perform isotopic analyses for iodine is a failure to meet TS Surveillance Requirement 4.4.5 (Violation 50-397/9603-02).

The failure to take the sample had elements of three recurring weaknesses: failure to follow procedures; poor log-keeping; and weakness

in communications, command, and control. The failure to utilize PPM 3.2.1 as an integrated part of the shutdown also contributed to the above violation. The on-duty crew during the downpower evolution which resulted in the 17 percent power change did not log the magnitude of the power change and did not realize that the chemistry sample was required. Had an appropriate log entry been provided, the oncoming crew may have been able to recognize the need for the sample.

The licensee disagreed that the failure to take and perform a primary coolant analyses was a violation of TS Surveillance Requirement 4.4.5. The licensee contends that the sampling specified in Table 4.4.5-1 and Action c are only required when the TS Limiting Condition for Operation 3.4.5 on activity limits are exceeded (meaning that no sampling was required for a power level change of greater than 15 percent in 1 hour, unless activity limits are exceeded). In addition, the licensee contends that other licensee's have obtained TS clarifications which recognize this interpretation of the TS.

The inspector disagreed with the licensee's interpretation. The TS seems clear in requiring a sample after a power level change of greater than 15 percent in 1 hour. In addition, while other licensees may have obtained specified wording changes to their plant TS, such wording changes had not been made in WNP-2 TS. The inspector concluded that the licensee's failure to sample and perform an analyses constituted a failure to meet TS Surveillance Requirements.

The inspectors noted one other instance of weaknesses in procedural usage and command and control in the shutdown process. PPM 3.2.1 indicated that the procedural steps should be performed in sequence unless the CRS or shift manager (SM) authorized steps to be performed out of sequence to take into account the current plant configuration or conditions. The inspector noted that the CRS had not communicated to the crew that the procedure steps would not be performed in order. The inspector observed that the crew was not performing the procedure in sequence and the lead RO and SM did not appear to know that the procedure was supposed to be performed in sequence.

Main Generator Over-Excitation Alarm: On March 2, a main generator over-excitation alarm was received. Through interviews with the involved operators, the inspectors found that the operators did not know the value of the main generator over-excitation alarm setpoint. The inspectors also determined that procedural guidance in PPM 3.2.1 associated with making voltage adjustments was misleading. Additionally, upon researching this issue, the licensee found that the alarm setpoint had drifted to a value lower than the desired alarm setpoint. Collectively, these deficiencies contributed to the chain of events that created the alarm condition.

To maintain reactive load close to zero, operators adjusted generator output voltage. With each adjustment, the voltage approached the

generator over-excitation alarm; however, operators were unaware of this since they did not know the alarm setpoint. During a minor grid fluctuation, the generator over-excitation alarm was received and the operators responded by reducing the main generator output voltage.

The inspectors considered the guidance in PPM 3.2.1 misleading because Step 24 stated "adjust main generator MVARs [reactive load] to maintain proper bus voltage"; however, there was no reactive loads (MVAR) adjustment on the control room panel. The inspectors observed that generator output voltage did not decrease as generator load decreased because it was regulated and remained relatively constant. Although MVARs increased in the negative direction as load dropped, the anticipated MVAR values were well within the design parameters of the system and no adjustments of MVARs were actually necessary.

In responding to the inspectors' questions, the licensee identified that the main generator over-excitation alarm was received prior to the desired setpoint. The alarm was received at approximately 25,400 volts, which was below the setpoint of 26,250 volts. The licensee initiated actions to check the calibration of the associated instruments. The licensee planned to provide guidance to operators to preclude inadvertent actuation of the alarm in the future. The licensee also planned to change PPM 3.2.1 to clarify the noted procedural requirements.

RFT Exhaust Temperature High Alarm: On March 2, with the reactor at approximately 15 percent power, the inspectors observed an exhaust temperature high alarm for RFT A. Plant operators raised the RFT A speed from 500 to 1200 rpm, as directed by the alarm response procedure, and the alarm cleared. The inspectors noted that PPM 3.2.1, paragraph 5.1, Step 3, stated "Remove the second reactor feedwater pump from service per PPM 2.2.4." The step also noted "The RFT being removed from service may be left rotating at 800 rpm." The inspectors questioned the licensee on low rpm operation of the RFTs, as this appeared to be the cause of the exhaust temperature high alarm. The licensee indicated that it had been a practice to operate the out-of-service RFT at low rpm to mitigate the effects of a loss of the in-service RFT. The inspectors questioned several licensed operators concerning low rpm operation of RFTs. The operators were not aware that very low rpm of the RFTs would result in high exhaust temperatures and could adversely affect turbine performance. Additionally, some operators were not knowledgeable of the low rpm setpoint when operating with the controller in automatic. The inspector discussed low rpm operation with a turbine vendor and found that very low rpm turbine operation could result in reduced turbine life. The inspector discussed low rpm operation with the licensee system engineers. The system engineers were aware of the potential for high turbine exhaust temperatures with low rpm operation, but were not aware that plant operating procedures allowed extended low rpm operation of the RFT. The licensee stated they would reassess the low rpm operation of the RFTs. The inspector considered this issue to be an example of operators not

questioning alarms and abnormal operation of plant equipment and the lack of system engineer awareness of the occurrence and impact of low rpm RFT operation while shutting down.

Reactor Water Level Reduction Prior to Reactor Scram: On March 2, the inspectors observed ROs, per paragraph 5.4, Step 6 of PPM 3.2.1, lower reactor vessel water level from 36 to 25 inches just prior to manually scramming the reactor for shutdown. The inspectors observed that, during the transient from the reactor scram, water level decreased to 13 inches, further decreased to 3 inches, and then recovered to 28 inches. Operators entered Emergency Operating Procedure 5.1.1 when reactor vessel water level went below 13 inches. The inspectors questioned the licensee on this operating practice. The licensee indicated that the shutdown procedure had been revised in 1990 to incorporate the practice of lowering reactor water level prior to the scram from low reactor power levels because operators had experienced several high water level transients which resulted in RFT trips following manual reactor scrams at low power levels. This practice had become an operator workaround in that feedwater level control system problems had previously caused the high water level on a manual trip, but improvements in system performance and operator training had resolved the problem. The inspector reviewed the licensee's 10 CFR 50.59 screening evaluation associated with the 1990 procedure change. The inspector found the screening evaluation determined no 10 CFR 50.59 safety evaluation was required. The inspector's review identified that the FSAR indicated that reactor water level was automatically maintained at 36 inches and did not describe lowering water level during a reactor shutdown. The change to the shutdown procedure appeared to constitute a change to an activity described in the FSAR that would require a 10 CFR 50.59 safety evaluation. The inspector also questioned what effect this change may have had on nuclear safety. At the conclusion of this inspection, the licensee was evaluating the potential safety impact of this change. This issue is an unresolved item (Unresolved Item 50-397/9603-03). The inspector considered this to be an example of operators not questioning abnormal plant operation.

Oncoming ROs: The inspectors observed on March 1 that two ROs, who were scheduled to work the following shift, had started responding to alarms on the secondary panel without first receiving a complete turnover or briefing on plant status. The inspectors considered the participation of the oncoming ROs in the shutdown evolution, without first receiving an appropriate brief on plant status, to be an indicator of weaknesses in command and control of the oncoming ROs.

Management Oversight: The inspectors observed that the licensee had additional management oversight in the control room during the conduct of the reactor shutdown. However, the management oversight personnel had not noted the above concerns.

c. Conclusions

Compared to previous reactor shutdowns, the licensee's conduct of the reactor shutdown during this inspection period was generally better coordinated. Three-way communications and second person verifications were considered improved when compared to past performance. Several weaknesses were identified associated with procedural adherence and usage, resulting in a cited and a noncited violation. These examples included: (1) the failure to verify the shutdown procedure prior to use (self-identified, noncited violation); (2) the failure to perform a required chemistry sample of the reactor coolant (NRC-identified, cited violation); and (3) the failure to perform the shutdown procedure in sequence or provide specific management direction to the contrary.

Additionally, the inspectors identified several instances where operations did not exhibit a strong questioning attitude when unexpected alarms or unwarranted operating conditions occurred. The conditions included a main generator over-excitation alarm, a high turbine exhaust temperature alarm, and an entry into emergency operating procedures as a result of lowering reactor vessel water level from 36 to 25 inches prior to the reactor scram. A combination of confusing procedural guidance, inadequate operator knowledge, and out-of-tolerance equipment contributed to some of the alarm conditions. These weaknesses were not noted by licensee management oversight personnel.

01.3 Equipment Tagging (71707)

The inspectors observed selected equipment for which tagging requests had been initiated and verified that tags were in place and the equipment was in the condition specified. The inspectors identified several instances where the licensee was utilizing the tag-out process to preclude performing timely permanent plant modifications. These observations are discussed in Section E1.1.

02 Operational Status of Facilities and Equipment

02.1 Engineered Safety Feature (ESF) System Walkdowns (71707)

The inspectors used Inspection Procedure 71707 to walk down accessible portions of the following ESF systems:

- Reactor Core Isolation Cooling System
- Standby Gas Treatment Train B
- Diesel Generator Division 2
- Low Pressure Core Spray System

Equipment operability, material condition, and housekeeping were acceptable in all cases. Several minor discrepancies were brought to the

licensee's attention and were corrected. The inspectors identified no substantive concerns as a result of these walkdowns.

08 Miscellaneous Operations Issues (92901)

08.1 (Closed) Inspection Followup Item 50-397/9324-01: licensee corrective action improvement program. This item was previously reviewed by the inspector in NRC Inspection Report 50-397/94-14. The review concluded that, although substantive changes had been made to streamline the corrective action program and to increase its effectiveness, procedural noncompliances continued. The licensee's Quality Assurance organization had reached similar conclusions about the corrective action program prior to the inspectors' independent evaluation.

In 1993 and 1994, WNP-2 operational performance appeared to have a modest improving trend. However, in the first quarter of 1995, a number of events occurred which raised concern about the effectiveness of the licensee's corrective action program. Concurrent with these events, the licensee conducted a self-assessment and concluded that additional corrective action program improvements were required. As a result, the licensee initiated quality assurance review of 100 percent of all problem evaluation requests (PERs) and established the corrective action review board to assess the adequacy and completeness of corrective actions for PERs categorized as significant. Since these changes have been implemented, the inspectors have recognized a slowly improving trend in the corrective action process. Although repetitive events occur, the magnitude and significance of the events appear to have decreased.

In August 1995, an integrated assessment team evaluated the licensee's programs to identify issues, perform root cause analyses, and implement corrective actions. The team found that, while considerable resources had been spent on various initiatives, their effectiveness was not yet evident.

Additionally, the NRC established an Oversight Panel chartered to independently monitor and assess the effectiveness of the licensee's corrective action program to achieve and maintain sustained improved performance at WNP-2. The Oversight Panel will remain in effect until no longer deemed necessary by the Regional Administrator, Region IV, in consultation with the Director of the Office of Nuclear Reactor Regulation.

Based on the slowly improving trend of the licensee's corrective action program and the close monitoring of the corrective action program by the NRC Oversight Panel, this item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (61726, 62703)

The inspectors observed all or portions of the following work order tasks and surveillance activities:

- Work Order Task XL91-01: periodic (4 years) electrical maintenance, inspection and testing for 480V breaker MC-3B
- PPM 7.4.5.1.7: low pressure core spray system operability test
- PPM 7.4.2.1: average power range monitor adjustments for TS action statement 3.2.2
- PPM 7.4.4.6.1.1B: reactor pressure vessel cooldown surveillance (documentation review only)

b. Observations and Findings

The inspectors found the work performed under these activities to be professional and thorough. All work observed was performed with the work package present and in active use. In general, technicians and operations personnel were technically knowledgeable and worked in a systematic and controlled manner. No problems were identified.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Violation 50-397/9412-02: errors in PPM 2.10.3, "Control, Cable, and Switchgear Rooms HVAC," Revision 21, and control room chiller temperature instrument setpoints not set in accordance with PPM 2.10.3. The inspector verified the corrective actions described in the licensee's response letter dated May 27, 1994, to be reasonable and complete. The inspector identified no similar problems.

III. Engineering

E1 Conduct of Engineering

E1.1 Plant Modifications

a. Inspection Scope (37551)

The inspectors reviewed clearance orders to ensure that modifications were performed where appropriate and that 10 CFR 50.59 evaluations were completed when necessary.

b. Observations and Findings

Clearance Order (CO) Usage: During a sampling review of selected active clearances, the inspectors identified three instances where the licensee had utilized the CO process to permanently remove equipment from service, instead of performing formal modifications to the facility. The inspector found in each case that the licensee had initially intended to repair the equipment that had been tagged out of service, but subsequently decided not to repair the equipment. The COs were between 18 months and 5 1/2 years old. The inspector identified that the licensee had not revised the FSAR to reflect the actual plant configuration for any of the COs. The COs included:

- CO 90-9-64, "Troubleshoot Engine Run Problem," dated September 19, 1990, removed from service the diesel motor that powered one of the air compressors in the Diesel Generator 2 air start system. The inspector found that in August 1991 the licensee had decided that the motor would be permanently removed from service, as documented in a 10 CFR 50.59 evaluation, but did not change FSAR Section 9.5.6.2 to reflect the change to the facility.

- CO 93-10-0128, "Fan Guard is Bent into Fan Blade," dated October 10, 1993, removed from service the air conditioning unit that cooled the electronics room on the 522 foot level of the reactor building. The unit was originally removed from service in 1989 due to a failure of the condenser coil. The inspectors found that in January 1993 the licensee had decided to permanently remove the unit from service, as documented in a 10 CFR 50.59 evaluation, but did not update FSAR Section 9.4.2.2.4.d to reflect the change to the facility.

- CO 94-07-0279, "Electrohydraulic Operator for WOA-V-52D," dated July 23, 1994, removed the operator from the control room HVAC purge line isolation valve (WOA-V-52D). The FSAR stated that the valve had an electrohydraulic operator which was powered from a Class 1E power source.

The inspectors identified that in 1988 the licensee removed the Class 1E power source from the operator (a formal modification with a 10 CFR 50.59 evaluation), but did not update FSAR Section 6.4.4.1.

In response to the inspectors' concerns regarding the utilization of COs in lieu of permanent plant modifications, the licensee agreed to periodically review existing COs as part of the temporary modifications review process. This would ensure that COs would not be utilized to remove equipment from service when formal modifications are appropriate.

As identified above in three instances, the licensee failed to revise the FSAR as required by 10 CFR 50.71(e)(4) regarding changes to the facility as described in the FSAR. 10 CFR 50.71(e)(4) required the licensee to update the FSAR annually and the FSAR revisions were required to be accurate to within 6 months of filing the revisions. The inspectors noted that in the last 12 weeks they have identified five instances of not appropriately updating the FSAR, which may suggest that system engineers' knowledge of their assigned system(s), the FSAR, and what constitutes a change to the facility is not adequate. In discussing these examples, the licensee stated that their response to the Notice of Violation issued with NRC Inspection Report 50-397/96-02 would also incorporate corrective actions common to the violation identified in this report. The licensee has indicated in the Performance Enhancement Strategy that an FSAR upgrade program is underway with a completion date and submittal to the NRC by approximately August 1997. The inspectors concluded that these three instances of failing to update the FSAR were additional examples related to an earlier violation for which the licensee had not had a reasonable opportunity to implement corrective actions. This item will be reviewed as part of the closeout of the violation identified in NRC Inspection Report 50-397/96-02 and is considered an inspection followup item (IFI 50-397/9603-05).

c. Conclusions

The inspectors identified three examples associated with the failure to update the FSAR. In these three instances, the licensee had utilized the CO process to permanently remove installed plant equipment from service in lieu of performing (or completing) permanent modifications. Furthermore, in the last 12 weeks NRC inspectors have identified five instances of not appropriately updating the FSAR, which may suggest that the system engineers' knowledge of their assigned system(s), the FSAR, and what constitutes a change to the facility is not adequate.

E2 Engineering Support of Facilities and Equipment

E2.1 Position Indication for Valve WOA-V-52D

a. Inspection Scope (37551)

The inspectors reviewed the adequacy on the control room position indication for Valve WOA-V-52D, the valve discussed in Section E1.1.

b. Observations and Findings

The inspectors identified an additional concern regarding the adequacy and utilization of the control room position indication for Valve WOA-V-52D. When the valve operator was removed, the licensee gagged the valve open and the position indication limit switch was made to always indicate open using tie wraps. The limit switch linkages to valve stem were also removed. The position indication provided only limited information because it did not indicate actual valve position. As such, if the gag failed and the valve drifted closed, the change in valve position would go undetected because the position indication had been made inoperable. The inspectors found that some operators were not aware that the valve was gagged open or that the indication had been modified such that the valve would always indicate open.

The inspectors identified that an abnormal operating procedure (4.10.3.1) required verification of the valve position using the control room indicating lights. Operators indicated they would use the control room indication to meet the requirements of the abnormal operating procedure. The inspectors determined that due to the tie wrapped limit switch the only valid means of determining valve position was to locally verify the valve position. In response to the inspectors' concern, the licensee indicated that they would make appropriate changes to the CO procedure to ensure that, in these types of instances, a tag is placed in the proximity of the control room indicating lights to alert operators of limitations of the position indicators. Additionally, the licensee planned to reinstall linkages from the valve stem to the valve limit switches so that the control room indication would again show actual valve position.

c. Conclusions

The licensee's modification of the position indication circuitry for Valve WOA-V-52D did not consider the use of the valve position lights in procedures. The licensee has initiated corrective actions to resolve this issue.

E2.2 Testing of Charcoal from Atmosphere Cleanup Systems

a. Inspection Scope (37551)

The inspectors reviewed procedures and documents related to the testing of ESF charcoal used in ESF atmosphere cleanup systems. The review included assessments of past test results.

b. Observation and Findings

The standby gas treatment system and the control room ventilation system use impregnated charcoal to adsorb radioactive gases, thereby reducing the radioactivity of the effluent stream during accidents.

TS 4.6.5.3.b.2 (Standby Gas Treatment System) and TS 4.7.2.c.2 (Control Room Emergency Filtration) state "verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide (RG) 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of RG 1.52, Revision 2, March 1978, for methyl iodide penetration of less than 0.175 percent and 1 percent, respectively. RG 1.52, March 1978, Paragraph C.6.a.(2) and a.(3) refer to Table 2, which requires charcoal testing per Table 5-1 Item 5.b of ANSI N509-1976. This testing is done with methyl iodide at 80°C and 70 percent relative humidity."

The licensee's FSAR, page 6.5-8, states "all postdelivery and postoperation tests are performed to meet the objectives of RG 1.52, Revision 2 using ANSI N510-1980. At least once each 18 months . . . a test canister is removed . . . and sent off for testing. Each sample is tested with methyl iodide per ASTM D3803 (1980) Method B at 70 percent relative humidity as defined in RG 1.52, Revision 2 (using ANSI N510-1980)."

The NRC Safety Evaluation Report in paragraph 6.5.2 states that conformance to the acceptance criteria provides the bases for concluding that the ESF filter system meets the requirement of 10 CFR Part 50.

The licensee contracts the testing of the ESF activated charcoal. NCS Corporation, the contract laboratory, tests the charcoal of the standby gas treatment system and the control room ventilation system per ANSI N509/N510 and ASTM D3803 (1986), Method B, with the exception that the test is conducted at 70 percent relative humidity. The control room charcoal is pre-equilibrated to 30°C and the standby gas treatment charcoal is pre-equilibrated to 70°C.

Therefore, a discrepancy appears to exist between the FSAR and the TS, i.e., the TS would have testing performed in accordance with ANSI N509-1976 and the FSAR would have testing performed in accordance

with ANSI N510-1980. The difference is that ANSI N-509-1976 establishes testing the control room charcoal adsorber at 80°C while ANSI N510-1980 establishes testing of the control room charcoal adsorber at 30°C. The testing for the standby gas treatment system charcoal is the same in both standards. The resolution of the correct temperature for testing of the control room charcoal is an inspection followup item (50-397/9603-05).

The licensee's improved TS submittal has changed the wording of this TS. The improved TS (Section 5.5.7.c) states "demonstrate for each ESF system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52, Revision 2, Section C.6.b, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1986 . . ."

c. Conclusions

The inspectors concluded that the licensee is adhering to what they stated in the FSAR with respect to the testing of ESF charcoal adsorbent. The licensee is conducting testing of the charcoal using the more recent standard. The safety evaluation report review accepted the licensee's method of testing of the ESF charcoal adsorbers. In discussions with NRR, the inspectors concluded that this would be an open item pending further NRC review.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-397/9526-01: lack of timely initiation of PERs. This item addressed an NRC inspector's observation of untimely initiation of a PER following the licensee's identification of degraded snubbers. The snubbers were of a special design and were connected to the discs of the containment wetwell-to-drywell vacuum breaker valves. This item also referred to the lack of timely PER initiation as observed in NRC Inspection Report 50-397/95-201, an integrated assessment team report.

The inspector reviewed the observations, interviewed licensee personnel, and checked related documents, which evidenced two examples of a violation of Procedure PPM 1.3.12, "Problem Evaluation Report." Section 5.1 of the procedure required personnel identifying a problem which affected equipment relied upon in the plant design basis to initiate a PER in a timely manner "(typically less than one day)."

The first example, identified in NRC Inspection Report 50-397/95-26, involved degraded vacuum breaker snubbers which could have affected the operability of the primary containment. These vacuum breakers are provided to equalize the pressure between the wetwell and drywell in order to ensure the structural integrity of the primary containment under conditions of large differential pressures. During the week of July 17, 1995, engineers disassembled three potentially degraded snubbers. The

engineers concluded that two of the snubbers were degraded, because the metal retaining tab for the capstan spring of the torque transfer drum had bent and no longer fully restricted the movement of the capstan spring. On August 7, 1995, the licensee informed the resident inspector of the degraded snubbers and noted that the snubbers were being sent to a facility for additional testing. The resident inspector questioned the licensee whether a PER had been initiated; the licensee then initiated PER 295-0922.

The second example, identified in NRC Inspection Report 50-397/95-201, involved a torque head used to torque flange fasteners for a main steam safety/relief valve. The torque head could not be located to perform the required post-use calibration. This was a concern because the safety valve function of the safety/relief valve assembly might have been affected. The safety valve function serves to protect the integrity of the reactor coolant system from overpressure conditions. Although the discovery date was July 19, 1995, a PER was not signed by the originator until July 31, 1995, and the SM had not signed the PER until August 8, 1995.

The aforementioned examples of untimely PER issuance violated the procedural requirements of PPM 1.3.12 and, therefore, represent two examples of a violation of 10 CFR Part 50, Appendix B, Criterion V (Violation 50-397/9603-06).

The inspector noted that the licensee had already identified the problem of untimely PER resolution for the first example in PER 295-0931. Several corrective actions were proposed in the PER. The inspector verified the corrective actions described in the PER to be reasonable and complete to the extent that the corrective actions would adequately address both examples of the violation. The inspector considered that no further action by the licensee was necessary.

E8.2 <u>(Closed) Violation 50-397/9412-01</u>: failure to promptly restore control room Chiller CCH-CR-1B to an operable status. The inspector verified the corrective actions described in the licensee's response letter dated May 27, 1994, to be reasonable and complete. The inspector identified no similar problems.

E8.3 <u>(Closed) Unresolved Item 50-397/9351-03</u>: potential discrimination against employee who raised safety concerns. This item documented an instance of apparent discrimination against a Supply System employee, which occurred after the employee had raised a safety concern. The licensee investigated the apparent infraction of 10 CFR 50.7 and responded to the NRC in a letter dated June 2, 1995. The NRC reviewed the response and acknowledged (in a letter to the licensee dated January 3, 1996) that: (1) the licensee confirmed the individual had been subjected to discriminatory treatment; (2) the Supply System made every effort to settle the matter to the satisfaction of the complainant, and was successful in doing so; (3) the Supply System had taken actions

to address the overall environment for raising concerns; and (4) the corrective actions taken were acceptable. Consistent with Section VII.B.5 of the NRC Enforcement Policy, the NRC has decided that it will not pursue enforcement action.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiological Preparations for Refueling Outage

a. Inspection Scope (71750)

The inspectors observed several licensee dose reduction efforts. These efforts included installation of permanent and temporary shielding and the flushing of selected portions of very high radiation segments of systems ("hot spots") that contain radioactive fluid.

b. Observations and Findings

The inspectors observed selected portions of flushes that the licensee performed on residual heat removal (RHR) sensing lines and Valve RHR-V-80. The flushes were noted to be well planned and performed. Contact dose rates were reduced from 450 rem/hr to 500 mrem/hr and 4.5 rem/hr to 3 mrem/hr, respectively. Additionally, during this period the licensee performed flushes on other plant components such as Valves RWCU-V-136, RWCU-V-137, RFW-V-70, and RFW-V-71 and certain radioactive water drain lines. These flushes also significantly reduced contact and area radiation levels and should likely result in reduced exposures during upcoming outage maintenance activities.

The inspectors observed selected portions of the installation of permanent shielding in the RHR B pump room. The installation of the shielding was well planned and performed. The shielding reduced general area radiation levels in the RHR B pump room from 160 to 14 mrem/hr. Additionally, the inspectors noted that the licensee had installed shielding in the reactor water cleanup (RWCU) holdup pump rooms and the precoat tank room. The installation of the shielding in these areas also significantly reduced general area radiation levels.

c. Conclusions

The licensee continued to implement their Dose Rate Reduction Program to reduce general area radiation levels and thereby reduce total radiation exposures. Source term reduction efforts continue to be successful in reducing monthly radiation exposure.

R2 Status of Radiological Protection and Chemistry Facilities and Equipment

2.1 Process and Radiation Effluent Monitor (PRM-RE-1B) Calibration Error

a. Inspection Scope (92904)

The inspectors conducted initial followup of an issue when the licensee identified the intermediate range of the reactor building ventilation system gaseous radiation monitor, referred to as the stack monitor and designated PRM-RE-1B, had been incorrectly and nonconservatively calibrated. The inspectors discussed the event with licensee chemists and engineering personnel involved with the issue and verified the licensee's immediate corrective actions.

b. Observations and Findings

On March 6, 1996, the licensee notified the NRC, as required by 10 CFR 50.72(b)(1)(v), "Lost Emergency Assessment Capability," that Monitor PRM-RE-1B had been improperly calibrated and was, therefore, inoperable. Due to this error, the licensee determined that dose projections that would have been provided during accident conditions would have resulted in estimates that would be roughly a factor of 20 lower than actual dose values. The licensee determined that the monitor had been incorrectly calibrated since its installation in 1993. During their assessment of this issue, the licensee determined that the gas calibration factor had been incorrectly calculated because a value from the wrong scale of a log count rate meter had been used. The licensee's immediate corrective action was to determine a factor that would assure conservative dose projections.

The licensee initiated PER 296-0176 to investigate the issue and determine corrective actions to prevent recurrence and submitted a 14-day special report to the NRC in a letter dated March 20, 1996. The inspectors verified that the licensee had implemented the conservative dose projection factor. This issue is unresolved and will be further reviewed during a future radiological controls inspection (Unresolved Item 50-397/9603-07).

S1 Conduct of Security and Safeguards Activities

S1.1 Protected Area Entry Without Supervised Search

a. Inspection Scope (92904)

The inspectors conducted initial followup of an event where a licensee employee entered the protected area without a supervised search. The inspectors discussed the event with licensee security personnel and with selected personnel involved with the issue and verified the licensee's immediate corrective actions.

b. Observations and Findings

On March 8, 1996, in accordance with 10 CFR 50.72, the licensee notified the NRC that an employee had entered the protected area without a supervised search. Security management briefed the inspectors on the event, indicating that they had verified that only one employee, who had approved authorization for the protected area, had entered the protected area by going through the search train, but was not observed by security personnel. The licensee determined that the root cause of this event was personnel error in that the search officer had not advised his supervisor that he was leaving his post and the officer in the badge station had not ensured that there was an officer in the search area. The inspector verified that only one employee had accessed the protected area without a supervised search. The licensee initiated PER 296-0183 to investigate this issue and determine corrective actions to prevent recurrence. Licensee immediate corrective action included briefing all security crews on the expectations for informing the security supervisor prior to leaving their post.

This issue is described in Licensee Event Report 96-S01-00 and will be reviewed by the NRC as part of the evaluation of the licensee event report.

F1 Review of FSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the FSAR description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the FSAR description. While performing the inspection discussed in this report, the inspectors reviewed the applicable portions of the FSAR that related to the areas inspected. The following inconsistencies were noted between the wording of the FSAR and the plant practices, procedures, and/or parameters observed by inspectors. Three examples of the failure to update the FSAR were identified concerning removal of a diesel motor on the diesel generator air start system, removal of an air conditioning unit, and removal of power from a valve operator and then removal of the operator itself (see Section E1.1).

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on April 10, 1996. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

The licensee disagreed that a violation of TS Surveillance Requirements occurred for the performance of sampling and analyses of reactor coolant when a 17 percent power level change occurred in 1 hour. This is discussed in detail in Section 01.2.b.

PARTIAL LIST OF PERSONS CONTACTED

WNP-2

P. Bemis, Vice President for Nuclear Operations
T. Love, Chemistry Manager
W. Pfitzer, Compliance Specialist
C. Schwarz, Operations Manager
G. Smith, Plant General Manager
J. Swailes, Engineering Director
D. Swank, Regulatory and Industrial Affairs Manager
R. Webring, Vice President Operations Support

NRC

J. Clifford, Senior Project Manager, Office of Nuclear Reactor Regulation

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 61726: Surveillance Observations
IP 62703: Maintenance Observations
IP 71707: Plant Operations
IF 71750: Plant Support Activities
IP 92901: Followup - Plant Operations
IP 92902: Followup - Engineering
IP 92903: Followup - Maintenance
IP 92904: Followup - Plant Support

ITEMS OPENED AND CLOSED

Opened

50-397/9603-01	NCV	failure to document appropriate version of procedure
50-397/9603-02	VIO	failure to take required chemistry sample
50-397/9603-03	URI	reactor water level reduction prior to reactor scram
50-397/9603-04	IFI	failure to update the FSAR
50-397/9603-05	IFI	resolution of the correct temperature for testing of the control room charcoal
50-397/9603-06	VIO	failure to issue PERs in a timely manner
50-397/9603-07	URI	effluent monitor calibration error

Closed

50-397/9324-01	IFI	licensee corrective action improvement program
50-397/9351-03	URI	potential discrimination against employee who raised safety concerns
50-397/9412-01	VIO	failure to promptly restore control room chiller to an operable status
50-397/9412-02	VIO	errors in PPM 2.10.3, "Control, Cable, and Switchgear Rooms HVAC," Revision 21
50-397/9526-01	URI	lack of timely initiation of PERs

LIST OF ACRONYMS USED

CO	clearance order
CRS	control room supervisor
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
HVAC	heating, ventilation, and air conditioning
MVAR	reactive loads
NRC	U.S. Nuclear Regulatory Commission
PER	problem evaluation request3
PPM	plant procedure manual
RFT	reactor feedwater turbine
RG	Regulatory Guide
RHR	residual heat removal
RO	reactor operator
RWCU	reactor water cleanup
SM	shift manager
STA	shift technical advisor
TS	Technical Specifications
WNP-2	Washington Nuclear Project-2