ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Inspector(s):	S. A. Boynton, Senior Resident Inspector J. E. Spets, Resident Inspector G. W. Johnston, Senior Project Engineer
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ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/98-06

Operations

- Good command and control of the March 18 reactor startup and April 3 feedwater temperature reduction were evidenced by adequate planning, proper assignment of personnel responsibilities, clear communications, and a conservative approach to implementing the activities (Section O1.1).

- Control of plant equipment was generally effective in maintaining proper plant configuration. However, two examples were identified where a lack of understanding of the impact of plant configuration changes resulted in the failure to identify or address discrepancies between the configuration changes and the plant's licensing bases. Specifically, operators failed to (1) address the final safety analysis report requirement to drain the spray rings for freeze protection when the normal drain path was found to be inoperable, and (2) recognize entry into a Technical Specification (TS) action statement when the emergency cooling coils for control room air conditioning, Train A, were isolated for planned maintenance. The second example was identified as a violation of TS 5.4.1.a for failure to properly implement written procedures for control of plant equipment (Section O1.2).

Maintenance

- Although overall plant material condition remained good, the inspectors continued to find material condition deficiencies that had not been previously identified and tracked for resolution by the licensee. Deficiencies included leakage from components outside containment that could contain highly radioactive fluid following a loss-of-coolant accident, and a locked spring hanger on the residual heat removal system's minimum flow bypass line (Sections O2.1 and M2.1).

Engineering

- The licensee failed to recognize that operation of the residual heat removal system (RHR) with the minimum flow bypass valves closed in standby, constituted a change to the facility as described in the Final Safety Analysis Report (FSAR) in that the original FSAR showed the valves to be open. The licensee missed several opportunities to identify the need for a written safety evaluation to support the change. These included the development of original system operating procedures, a 1993 revision to the FSAR that changed the valves' position on the process data sheet, and the licensee's current FSAR upgrade project. A violation of 10 CFR 50.59(b)(1) and 10 CFR 50.9(a) was identified (Section E1.1).

- The licensee failed to provide adequate controls for the position of the RHR system's suppression pool return valves to ensure that during operations with the return valves open, the valves' position would be limited to meet the injection times assumed in the

loss-of-coolant-accident analyses. Specifically, with the return valves greater than approximately 40 percent open, full low pressure coolant injection flow to the vessel would not be achieved within the 66 seconds assumed by the analyses. The failure to adequately translate the design requirements of the RHR system to plant operating procedures and instructions was identified as a violation of 10 CFR Part 50, Appendix B, Criterion III (Design Control) (Section E1.1).

- Following the identification of leakage from the safety-related nitrogen supply to the automatic depressurization system (ADS), engineering personnel established an adequate technical basis for system operability. However, the technical justification was not appropriately documented in the associated problem evaluation request. Additionally, the problem evaluation request was closed without addressing the root cause of the degraded condition (Section E1.2).

Plant Support

- Inconsistent expectations for implementing radiological controls requirements resulted in several procedure noncompliances during an instrumentation and controls surveillance performed in a posted high radiation area. Specifically, an improper radiation work permit (RWP) was utilized for the job, and positive access control to the high radiation area was not maintained by a qualified health physics (HP) technician. Although the actual and potential dose consequences of the event were considered to be low, the generic implications were considered significant in that several administrative barriers to personnel overexposure were not properly implemented. A violation of TS 5.4.1.a was identified for failure to properly implement written procedures for radiation protection (Section R1.1).

Report Details

Summary of Plant Status

The plant began the inspection period in Mode 4, in an unplanned outage, due to the inadvertent closure of a main steam isolation valve (the event is discussed in NRC Special Inspection Report 50-397/98-05). The failed instrument air line to the valve's actuator was repaired and the plant reentered Mode 2 on March 18. The plant was returned to full power on March 22. Due to end-of-cycle fuel burnup, the licensee implemented final feedwater temperature reduction on April 3 to maintain 100 percent thermal power.

On April 18, the plant was placed in Mode 3 in preparation for its 13th refueling outage, scheduled for 53 days. The plant was in Mode 5 with core alterations in progress at the end of the inspection period. Outage activities will be documented in NRC Inspection Report 50-397/98-09.

I. Operations

O1 Conduct of Operations

O1.1 Plant Startup and Final Feedwater Temperature Reduction

a. Inspection Scope (71707)

The inspectors observed portions of the reactor startup on March 17-18 and the implementation of final feedwater temperature reduction on April 3.

b. Observations and Findings

Evaluation of the licensee's readiness for plant restart on March 18 was documented in NRC Special Inspection Report 50-397/98-05.

Pre-evolution and periodic briefings were appropriate in providing an adequate level of detail to identify critical tasks in the startup and personnel responsibilities. During the approach to criticality, the control room staff properly implemented the requirements of Plant Procedures Manual (PPM) 1.3.1, "WNP-2 Operating Policies, Programs, and Practices," Revision 36, and PPM 1.3.59, "Reactivity Management Program," Revision 4. Although there were a number of individuals participating in the startup for training purposes, training activities were properly monitored and controlled by the on-shift crew. The inspector noted that the area in front of the full core display, where the approach to criticality was being made, was clear of extra personnel, and the noise level was normal and not distracting. Three-way communications were utilized consistently during the startup.

During the approach to criticality, control rod manipulations were prolonged due to the need for the control room operators to periodically increase control rod drive header pressure to facilitate movement of the rods. However, the frequent changes in drive

header pressure did not adversely impact the operators' focus on control rod manipulations. The increased drive header pressure was necessary to overcome air entrapment in the control rod drive hydraulic (CRDH) system. The entrapment of air in the system is a known phenomenon to the licensee and is a result of the aeration of condensate water when main condenser vacuum is not maintained. Condensate is the normal supply to the CRDH system. From discussions with the system engineer, the control rods are normally exercised and/or tested for scram timing during planned outages, which eliminates or significantly reduces air entrapment in CRDH prior to startup. A lack of control rod movements during the forced outage between March 11-17 resulted in a larger amount of air in the CRDH system. The system engineer indicated that options were being evaluated to minimize the air entrapment during potential future unplanned outages.

On April 3, the licensee initiated a reduction in the final feedwater temperature to increase thermal power back to 100 percent. Power had slowly reduced to approximately 98 percent due to end-of-cycle coastdown from fuel burnup. The feedwater temperature reduction was accomplished by opening the bypass valve around the sixth string of feedwater heaters, which reduced feedwater temperature by approximately 15°F. The pre-evolution briefing was thorough and addressed personnel responsibilities for both normal and abnormal conditions during the evolution. Implementation was properly calculated to minimize the reactivity excursion from the temperature reduction.

c. Conclusions

Good command and control of the March 18 reactor startup and April 3 feedwater temperature reduction were evidenced by adequate planning, proper assignment of personnel responsibilities, clear communications, and a conservative approach to implementing the activities.

O1.2 Plant Configuration Control

a. Inspection Scope (71707)

As part of routine plant tours and review of plant logs, the inspector verified proper implementation of the licensee's configuration controls for safety-related structures, systems, and components.

b. Observations and Findings

Configuration controls for equipment required by TS were effectively implemented during the inspection period, with two notable exceptions.

On March 16, the operating crew noted in the control room log that the drain valves for both of the ultimate heat sink spray rings (Valves SW-V-171A and B) were stuck in the closed position. This material condition deficiency prevented operators from draining the

spray rings when spray operations were secured. The inspector noted that Section 9.2.5.2 of the FSAR requires the spray rings be drained for freeze protection during freezing weather, when not in operation. In discussing the condition of Valves SW-V-171A and -B with the operating crew, it was evident that the crew was unaware of the FSAR requirement and had not established an alternate method for draining the spray rings. Because the plant was still operating under potential cold weather conditions (defined by procedure as November 1 through April 1), the control room supervisor concluded that a temporary change to the service water system operating procedure was required to provide an alternate means to drain the spray ring. The inspector reviewed the procedure changes and found them adequate to address the FSAR requirement until the drain valves could be repaired.

On April 3, during a tour of the radwaste building, the inspectors noted that the service water outlet valves from the emergency cooling coils of the control room air conditioning (AC) Subsystem A were danger-tagged closed. A review of control room logs earlier had noted that the coils' emergency chill water supply was also unavailable due to planned maintenance on the chiller. The control room AC Subsystem B was operable.

WNP-2 TS 3.7.4 requires two control room AC subsystems to be operable while in Modes 1, 2, and 3. In accordance with the Bases of TS 3.7.4, the emergency cooling coils are required to be operable to meet the limiting condition for operation. The Bases further state that the coils may be supplied with either service water or emergency chill water. Without the availability of either cooling medium to the control room AC Subsystem A, the inspector questioned the control room supervisor (CRS) on the need to enter the actions of TS 3.7.4. The CRS agreed that Subsystem A was inoperable and documented the entry into the action statement of TS 3.7.4 in the Technical Specifications/Licensee Controlled Specification Surveillance Inop./LCO/RFO log. TS 3.7.4 requires the Subsystem A to be returned to service within 30 days. Problem Evaluation Request (PER) 298-0305 was initiated to document the failure to enter the action statement of TS 3.7.4 when the emergency cooling coils were removed from service on April 2.

PPM 1.3.1, "WNP-2 Operating Policies, Programs, and Practices," Revision 36, describes the process by which the operations department will control and track the condition of equipment required by TS and Licensee Controlled Specifications (LCS). Step 4.1.5 of PPM 1.3.1 states that when preventive maintenance renders a system or component inoperable per TS or LCS, the CRS/shift manager shall declare the associated system or component inoperable and enter it on the TS/LCSSurveillance Inop./LCO/RFO log. To assist the CRS/shift manager in implementing TS and LCS requirements, maintenance work orders that render TS or LCS required equipment inoperable are expected to include a worksheet (VET) describing the need and justification for voluntarily entering a TS or LCS action statement. The work order that removed the Subsystem A emergency cooling coils from service on April 2 did include a VET worksheet. However, when the work order was presented to the CRS, a hard copy of the worksheet was not available, contrary to normal practice. Furthermore, although

the worksheet was available as an electronic file on the local area network, the CRS was unable to access the file from the control room. Without the actual VET worksheet available for review, the CRS inappropriately utilized another worksheet that had been prepared for ongoing work on control room AC Subsystem A. That worksheet only addressed operability of the emergency chill water supply to the cooling coils under LCS and did not adequately address operability of the control room AC Subsystem A under TS. The failure of the CRS to follow the requirements of PPM 1.3.1 and to recognize and declare the control room AC Subsystem A inoperable per TS 3.7.4 was identified as a violation of TS 5.4.1.a (VIO 50-397/98006-01).

c. Conclusions

Control of plant equipment was generally effective in maintaining proper plant configuration. However, two examples were identified where a lack of understanding of the impact of plant configuration changes resulted in the failure to identify or address discrepancies between the configuration changes and the plant's licensing bases. Specifically, operators failed to: (1) recognize that inoperable drain valves for the service water spray rings were required by the FSAR for freeze protection; and (2) recognize entry into a TS action statement when the emergency cooling coils for control room AC, Train A, were isolated for planned maintenance. The second example was identified as a violation of TS 5.4.1.a for failure to properly implement written procedures for control of plant equipment

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns (71707)

a. Inspection Scope

The inspectors walked down accessible portions of the following engineered safety feature systems:

- Standby Liquid Control
- Control Room Emergency AC/Filtration System, Trains A and B
- Standby Service Water System, Train A
- Low Pressure Core Spray
- Residual Heat Removal (RHR), Trains A and B

b. Observations and Findings

The systems were generally found to be properly configured for the plant operating conditions. However, several problems were identified with configuration management of the standby service water system and control room AC system (See Section O1.2).

On March 15, while RHR Train A was in service for shutdown cooling, the inspector identified a constant-load spring hanger with its hydrostatic test locks engaged. The

hanger supported the minimum flow bypass piping between the RHR A injection line and Valve RHR-V-64A (RHR A Minimum Flow Bypass). The engagement of the locks caused the support to act as a rigid restraint to pipe motion. The licensee initiated PER 298-0232 to document the condition for evaluation and corrective action.

A conservative stress analysis performed by the licensee showed the potential for the piping to exceed ASME Code allowable stress limits by approximately 15 percent. The calculated stress was within all other ASME Code design limits, including faulted system loading. The licensee supplemented its analysis with nondestructive examinations of the four highest loaded pipe welds around the hanger. No observable damage or degradation was noted. A walkdown of all the safety-related constant load spring hangers in the reactor building was conduct by plant engineering personnel. Additional sampling was independently performed by the inspector. No other supports were found with their hydrostatic test locks engaged. The inspector did not identify any historical performance issues with the improper locking of spring supports.

Through the licensee's inservice inspection program, the support was last inspected in 1989. The licensee was unable to identify a root cause for the engagement of the hydrostatic test locks. However, the licensee concluded that the locks had likely been engaged for at least one cycle. To ensure the locks are not engaged inadvertently, the licensee applied caution labels to the spring hanger.

Due to the length of time that the locks were likely engaged, the inspector concluded that routine system and plant walkdowns by the licensee have not been effective in identifying this type of deficiency.

c. Conclusions

A spring hanger found with its hydrostatic test locks engaged indicated that plant walkdowns by licensee personnel have not been effective in identifying this type of deficiency. The safety consequences of the condition were determined to be low.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707, 61726)

The inspectors observed and/or reviewed the following work activities:

- OSP-ELEC-M702, Revision 2, Diesel Generator 2 - Monthly Operability Test
- OSP-INST-H101, Revision 8, Shift and Daily Instrument Checks (Modes 1, 2, 3)

b. Observations and Findings

The monthly surveillance on the Division II emergency diesel generator was properly implemented by the equipment operator in the emergency diesel generator room and involvement by the system engineer was also notable.

Review of OSP-INST-H101, performed on April 3, noted that containment hydrogen/oxygen monitoring instrumentation, Train A, was inoperable, leaving Train B to monitor both the wetwell and the drywell. A footnote in Procedure OSP-INST-H101 called for the Train A hydrogen/oxygen monitor to be normally aligned to the drywell following instrument readings, while Train B would be left aligned to the wetwell. The inspector noted that Train B was aligned to the drywell and questioned the shift manager on the configuration. The shift manager justified the configuration based upon the requirement to monitor drywell hydrogen and oxygen following a design basis accident, but agreed that the procedure did not adequately provide flexibility to allow it. A temporary change was subsequently implemented to allow the operable train to be aligned to the drywell when only one train is available.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Material Condition of Safety-Related Systems

a. Inspection Scope (37551)

The inspectors performed frequent plant tours and reviewed licensee logs and PERs to evaluate material condition of plant structures, systems, and components (SSCs).

b. Observations and Findings

In general, the inspectors noted that the material condition of safety-related SSCs, including reliability and availability, was very good. However, several equipment problems and material condition deficiencies were noted.

- The inspector identified leakage from the following SSCs that had not been previously identified or characterized by the licensee:

  - Valve RHR-V-24A (RHR Loop A Suppression Pool Return Valve) - historical leakage from the valve's packing was identified by standing water in the well of the bonnet and was noted in NRC Inspection Report 50-397/97-18. Due to increased use of the RHR system in suppression pool cooling (SPC), which cycled RHR-V-24A, the inspector noted that active leakage of 20-25 drops per minute had developed from the valve's packing.

- Valve LPCS-V-5 (Low Pressure Core Spray Injection Valve) - standing water on the valve's bonnet and stains on the floor beneath the valve indicated long-term, minor leakage through the valve's packing. The inspectors' identification of this deficiency was significant because the valve is readily observable at a standing height from the floor, and observation does not require entry into a contamination or high radiation area.

- RCIC Turbine Casing Flange - during surveillance testing of the reactor core isolation cooling system, the inspector identified packing leakage from RCIC-V-45 and a small steam plume from the flange interface between the upper and lower turbine casings. The casing leak had not been previously identified by the licensee. Although the V-45 valve packing leakage had been previously noted by the system engineer, a deficiency tag had not been hung to identify the leak, contrary to management expectations.

- The licensee identified that the drain valves for the standby service water spray rings were stuck shut. The deficiency was noteworthy in that the inability to open the valves precluded normal draining of the spray rings for freeze protection when they were not in operation. It also points to weaknesses in the maintenance program for these valves. See Section O1.2 for further discussion of this issue.

- Several component failures on Containment Hydrogen/Oxygen Monitor, Train A, resulted in unavailability of this redundant train for approximately 23 days. The failures included the sample pump and the solenoid operated sample isolation valve for the wetwell. The hydrogen/oxygen monitoring instrumentation has experienced several other failures in the past two years.

Each of the above SSCs is addressed in plant TS. The SSCs where leakage was identified by the inspector are all covered under the licensee's leakage surveillance and prevention program. That program is designed to identify and reduce leakage from systems outside containment that may contain highly radioactive fluid following a loss of coolant accident. Equipment operator rounds and tours by plant supervision could have provided additional opportunities for these leaks to be identified.

c. Conclusions

Although overall plant material condition remained good, the inspectors continued to find material condition deficiencies that had not been previously identified and tracked for resolution by the licensee. Deficiencies included leakage from components outside containment that could contain highly radioactive fluid following a loss-of-coolant accident.

III. Engineering

E1 Conduct of Engineering

E1.1 Operability of the RHR System During Nonstandby Lineups

a. Inspection Scope

The inspector reviewed the licensee's operation of the RHR system in various modes, other than standby, to determine if the operation was consistent with the design and licensing bases for the emergency core cooling system (ECCS).

b. Observations and Findings

Section 5.4.7 of the FSAR describes the design and operation of the RHR system. Section 6.3 of the FSAR also describes the ECCS, or low pressure coolant injection (LPCI) function of the system.

Section 5.4.7.1.1 of the FSAR outlines the 8 primary design operating modes of the RHR system. These include:

- LPCI Mode
- SPC and Containment Spray Cooling Modes
- Shutdown Cooling Mode
- Alternate Shutdown Cooling Mode
- Fuel Pool Cooling Mode
- Minimum Flow Bypass Mode
- Standby Mode
- Reactor Steam Condensing Mode

FSAR Discrepancies: The LPCI mode of RHR operates in conjunction the other ECCSs to provide adequate core cooling for all design basis loss-of-coolant accident (LOCA) conditions. The SPC and containment spray cooling modes are designed to provide cooling to maintain containment and suppression pool temperatures and pressures following major transients. FSAR Section 5.4.7.1.1.g states that during normal power operation when the RHR system is required to be available for the LPCI mode in the event a LOCA occurs, the system is required to be maintained in the standby mode. In this mode, the system is stated to be aligned with the pumps' suction from the suppression pool and the minimum flow bypass valves open, and all other RHR system valves aligned so that only the inboard LPCI injection valves are required to open and the RHR pumps started for LPCI flow to be delivered to the reactor.

Section 6.3 (ECCS) of the FSAR contradicts Section 5.4.7.1.1. Specifically, Section 6.3.2.2.4 states that the inboard LPCI injection valve is a testable check valve that opens on the motive force of the LPCI flow and that the outboard LPCI injection valve, an 18-inch motor-operated gate valve, must reposition. A review of the licensee's

LOCA analysis confirmed that a delay in injection flow is assumed for opening of the outboard injection valve. The actual operation of the LPCI system is consistent with Section 6.3, therefore, the statement in Section 5.4.7.1.1 appears to be in error.

Figure 5.4-14c, "RHR System Process Data," also provides information on the RHR system configuration in its various modes of operation, including the positions of major system valves. During initial plant licensing, in 1984, Figure 5.4-14c showed the minimum flow bypass valves to be opened in Standby. Plant Procedures Manual (PPM) 2.4.2, "RHR System," Revision 31, provides guidelines for operating the RHR system in its various modes of operation. In placing an RHR loop in the Standby mode, PPM 2.4.2 requires verification that the minimum flow valve is closed. In fact, each of the 31 revisions to the system operating procedure provided direction to close the valve when placing the system in Standby. As noted above, FSAR Section 5.4.7.1.1.g states that the minimum flow bypass valves are open in the Standby mode. This discrepancy between the operating procedure and the RHR system discription in the FSAR had not been evaluated by the licensee, in accordance with 10 CFR 50.59.

A review of plant records found that the licensee has had several opportunities to identify and disposition the discrepancy. In 1984, a design change was initiated to include logic relays in the minimum flow valve control circuit to close the minimum flow valve when its associated RHR pump breaker is open. In evaluating the change, the licensee failed to recognize that Figure 5.4-14c showed the valve open when the pump was off and the system in Standby. In fact, the licensee's justification for not performing a 10 CFR 50.59 evaluation included the statement that the change was "necessary to meet the FSAR commitments" in Section 5.4. No basis for that statement could be found. In 1993, the licensee revised Figure 5.4-14c to show that the minimum flow bypass valves are closed in standby. The change was made to reflect the operation of RHR, as described in PPM 2.4.2 and to correct what was termed a "drawing discrepancy that has been in place since 1983." Again, no evaluation was performed for the change to the figure.

Neither of the above errors was identified during the two independent reviews established by the licensee's FSAR upgrade program. Although the position of the minimum flow valves was questioned during an additional review of Section 5.4.7 that was performed concurrently with the inspectors, the licensee concluded that the description in Section 5.4.7.1.1 was merely in error and would be corrected through its FSAR upgrade program. The justification was similar to that used in the 1993 change to Figure 5.4-14c and a safety evaluation was not performed. The failure to document a written safety evaluation that provided the bases that the change in the position of the minimum flow bypass valves did not represent an unreviewed safety question was identified as a violation of 10 CFR 50.59(b)(1). The inaccuracies in the FSAR regarding the standby position of the minimum flow bypass valves were also identified as a violation of 10 CFR 50.9(a) (VIO 50-397/98006-02).

RHR Design Control: Based upon the licensee's current licensing basis LOCA analysis, injection flow from the operable RHR loop(s) is assumed to begin approximately 66 seconds from initiation of the event (38 seconds to reach injection

valve open permissive pressure setpoint, plus 28 seconds for injection valve stroke time). These times assume the system is initially in the Standby mode. Because the RHR systems were not analyzed for LPCI initiation while in other modes of operation, the inspector reviewed the licensee's controls for those other modes to ensure that either system operability was maintained, or that appropriate actions were taken in accordance with TS.

PPM 2.4.2 provides directions to operators for placing RHR Systems A and/or B in the SPC mode, and RHR System C in a "suppression pool mixing" mode. Although these modes are normally utilized for system testing, the licensee has expanded their use for cooling the suppression pool to remove heat added by leaking safety relief valves and for events where there is a loss of the associated keep-fill system. In initiating the SPC mode, PPM 2.4.2 directs operators to open Valve RHR-V-24A(B), SPC/Test Return, to obtain a desired flow between 7000-7500 gpm. The most recent operability surveillances performed for RHR Systems A and B indicated that the maximum flow achieved with Valve RHR-V-24A(B) fully open is approximately 7400 gpm. Based upon the range of flow rates provided in PPM 2.4.2 and the throttling characteristics of the return valves, the procedure would allow the position of the return valves to be between approximately 25 and 100 percent open while in the SPC mode. In establishing the full flow test or suppression pool mixing mode of RHR System C, PPM 2.4.2 did not provide a required flow rate or band. Therefore, it could be assumed that the C test return valve, Valve RHR-V-21, would be throttled between some minimum value and full open.

The nominal closing stroke time for each of the test return valves is approximately 88 seconds, with a maximum allowable stroke time of 114 seconds. Assuming the return valve is fully open during system operations as described above, while applying the same timing assumptions utilized in the licensee's LOCA analysis, the realignment of the RHR system from the SPC or full flow test mode to the LPCI mode could take up to 132 seconds (3 seconds for instrument delay, 15 seconds for diesel startup, and 114 seconds for the closing stroke time of the return valve). This would result in full injection flow being achieved approximately 66 seconds later than assumed in the LOCA analysis. Based upon the closing speed of the RHR return valves, the inspector concluded that the valves would have to be maintained less than approximately 40 percent open to ensure that the assumed injection times in the LOCA analysis were met. Procedure PPM 2.4.2 did not provide specific controls to meet this criteria and the licensee considered the RHR systems operable when aligned in these modes. Due to a lack of historical data, the inspector could not identify any periods of time, with the exception of testing, where the position of the suppression pool return valves placed the RHR systems outside of their design.

During TS required quarterly surveillances of the RHR subsystems, the suppression pool return valves are fully opened for stroke time testing. As discussed above, however, the licensee failed to recognize that while the return valves were full open for testing, the system was outside of its design and, therefore, inoperable. The failure of the RHR system operating and surveillance procedures to control the position of the suppression

pool return valves to within the limits of the licensing basis LOCA analysis was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" (VIO 50-397/98006-03).

As an immediate corrective action, the licensee revised procedures to ensure that operation of the RHR suppression pool return valves maintains the RHR systems within design requirements. The revisions were found to adequately address the concerns.

c. Conclusions

The licensee failed to recognize that operation of the RHR system, with the minimum flow bypass valves closed in standby, constituted a change to the facility as described in the FSAR in that the original FSAR showed the valves to be open. The licensee missed several opportunities to identify the need for a written safety evaluation to support the change. These included the development of original system operating procedures, a 1993 revision to the FSAR that changed the valves' position on the process data sheet, and the licensee's current FSAR upgrade project. A violation of 10 CFR 50.59(b)(1) and 10 CFR 50.9 was identified.

The licensee failed to provide adequate procedural controls for the position of the RHR system's suppression pool return valves to ensure that during operations with the return valves open, the valves' position would be limited to meet the injection times assumed in the LOCA analyses. Specifically, with the return valves greater than approximately 40 percent open, full LPCI flow to the vessel would not be achieved within the 66 seconds assumed by the analyses. The failure to adequately translate the design requirements of the RHR system to plant operating procedures and instructions was identified as a violation of 10 CFR Part 50, Appendix B, Criterion III (Design Control).

E1.2 Leakage from the Containment Instrument Air (CIA) System

a. Inspection Scope

Prior to the plant startup from the March 11 forced outage, the licensee identified an increase in nitrogen usage for the CIA system. The inspector reviewed the licensee's actions to both identify and correct the leakage of nitrogen and prevent recurrence.

b. Observations and Findings

As noted in NRC Inspection Report 50-397/98-05, the CIA system engineer appropriately questioned an unusually high nitrogen flow rate observed during a routine system walkdown. The flow rate of approximately 1.8 scfm was slightly elevated from a normal range of 0.7 to 1.6 scfm. PER 298-0237 was initiated to document the concern. From its planned troubleshooting efforts, the licensee found that the increased flow rate was attributable to leakage past the valve seats of Valves CIA-RV-5A and 6A. These valves provide for pressure relief on the ADS Train A nitrogen supply header and the nonsafety-related supply header respectively. The leakage past Valve CIA-RV-5A (approximately

0.3 scfm) was significant in that it exceeded the allowable leakage from ADS Train A (.07 scfm). The limit for allowable leakage is established to ensure that there is a 30-day supply of nitrogen to the ADS valves to meet system design requirements.

The licensee manually lifted and reseated both of the relief valves and was successful in reducing the overall leakage to "small amounts," as described in PER 298-0237. From subsequent discussions with the engineering supervisor who dispositioned the PER, the inspector determined that the licensee used engineering judgement to conclude that the results of the qualitative as-left testing performed on the relief valve were within the allowable leakage limit. Based upon those results and the fact that the nitrogen flow rate had returned to a level consistent with its historical operating range, the licensee determined that the system was operable. Although the inspector found that the licensee's technical arguments for system operability were sound, the resolution of PER 298-0237 did not adequately document those arguments to provide a technical basis for concluding that the as-left leakage from the system was within the allowable leakage rate of .07 scfm. The licensee subsequently revised the PER to provide the documented justification.

The inspector also noted that although PER 298-0237 addressed the immediate concern of unacceptable system leakage, the PER was closed without addressing the root cause of the leakage past the relief valve seats. The system engineer indicated that the relief valves had likely lifted during system restoration from outage work and noted that the root cause of the lifting would be evaluated. However, without a PER to track the resolution, it was unclear as to what vehicle would be utilized for performing the evaluation. Subsequently, a corrective action was assigned to the PER to evaluate potential weaknesses in the system restoration procedure.

c. Conclusions

Following the identification of leakage from the safety-related nitrogen supply to the ADS, engineering personnel established an adequate technical basis for system operability. However, the technical justification was not appropriately documented in the associated PER. Additionally, the PER was closed without addressing the root cause of the degraded condition.

IV. Plant Support

R1.1 Inconsistent Expectations for Implementing Radiological Controls

a. Inspection Scope (71750)

The inspector reviewed the radiological controls established for a surveillance on the main condenser low vacuum pressure switches, MS-PS-56A and B.

b. Observations and Findings

During a plant tour on April 3, the inspector observed an instrumentation and controls (I&C) technician performing a surveillance within a posted high-high radiation area. Specifically, the technician was calibrating the main condenser low vacuum pressure switches, MS-PS-56A and B. These switches are located within the north labyrinth entry of the turbine deck, on the 501-foot elevation of the turbine building. While the plant is operating at full power, the entrance to the labyrinth is normally locked, in accordance with TS, due to radiation levels on the turbine deck being greater than 1000 mR/hr. The radiation levels within the labyrinth itself are generally less than 100 mR/hr.

The inspector noted that continuous coverage by a HP qualified individual was not being maintained while the door to the labyrinth was unlocked. The HP technician who was covering the job had left the turbine building to survey swipes taken in the work area for loose surface contamination. From subsequent conversations with the HP staff, the inspector determined that the HP technician had relinquished positive access control of the high-high radiation area to the I&C technician during his absence. The HP staff indicated that this was an accepted practice. PPM 11.2.7.3, "High and Very High Radiation Area Controls, "Revision 15, outlines the radiological controls requirements for accessing high-high radiation areas. Section 5.2.8.e of PPM 11.2.7.3 states that any time a high-high radiation area door has been unlocked, an HP qualified individual shall provide positive access control of the area while the door is unlocked. Section 4.6 notes that the term "HP qualified individual" identifies an individual who is qualified in health physics procedures (e.g., WNP-2 HP technicians and qualified contractor HP technicians). Therefore, the inspector concluded that the practice of relinquishing positive access control to non-HP qualified individuals was inconsistent with plant procedures. The failure to maintain positive access control of the unlocked high-high radiation area door to the turbine deck by an HP qualified individual was identified as a violation of PPM 11.2.7.3 and the first example of a violation of TS 5.4.1.a (VIO 50-397/98006-02).

The work orders associated with the I&C surveillance utilized RWP 98000162 for establishing the radiological controls for the job. RWP 98000162 is a group RWP for plant maintenance tasks expected to total less than 30 mrem per task. The special instructions of the RWP noted that entry into high and high-high radiation areas is not authorized. The dose rates in the labyrinth were not included in the RWP. Section 5.2.4 of PPM 11.2.7.3 states that access to each such area shall be controlled by means of an RWP that includes dose rates in the immediate area. The root cause of using an improper RWP appeared to result from inconsistent implementation of radiological controls for the recurring task. Specifically, the radiation protection manager indicated that on other occasions the labyrinth was deposted and the high-high radiation area boundary was relocated to the other entrance to the labyrinth. If properly deposted, RWP 98000162 would have been appropriate for the task. The failure to provide

information on the dose rates in the labyrinth in RWP 98000162 was identified as a violation of PPM 11.2.7.3 and the second example of a violation of TS 5.4.1.a (VIO 50-397/98006-04).

The total dose received by personnel during this particular activity was less than 20 mrem. Noting that the work area was limited to the labyrinth, and did not require ingress to the turbine deck, the inspector concluded that the safety significance was low in relation to the actual and potential dose consequences for the specific evolution.

c. Conclusions

Inconsistent expectations for implementing radiological controls requirements resulted in several procedure noncompliances during an instrumentation and controls surveillance performed in a posted high radiation area. Specifically, an improper RWP was utilized for the job, and positive access control to the high radiation area was not maintained by a qualified HP technician. Although the actual and potential dose consequences of the event were considered to be low, the generic implications were considered significant in that several administrative barriers to personnel overexposure were not properly implemented.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on April 23, 1998. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection could be considered proprietary. The licensee's LOCA analyses were identified as proprietary from the vendors who performed them. Details of the methodologies utilized to perform the analyses are not discussed in this report and the analyses were returned to the licensee.

ATTACHMENT

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Coleman, Regulatory Affairs Manager
D. Giroux, System Engineering
D. Hillyer, Radiation Protection Manager
T. Hoyle, Engineering Programs
D. Kobus, Fire Protection
A. Langdon, Assistant Operations Manager
P. Inserra, Licensing Manager
S. Oxenford, Operations Manager
G. Smith, Plant General Manager
J. Kane, Acting Engineering Manager
R. Webring, Vice President Operations Support

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

VIO 50-397/98006-01	failure of the CRS to recognize and declare the control room AC Subsystem A inoperable per TS 3.7.4
VIO 50-397/98006-02	failure to maintain positive access control of the unlocked high-high radiation area door to the turbine deck by an HP qualified individual
VIO 50-397/98006-03	failure of the RHR system operating and surveillance procedures to control the position of the suppression pool return valves to within the limits of the licensing basis LOCA analysis
VIO 50-397/98006-04	failure to provide information on the dose rates in the labyrinth in RWP 98000162

LIST OF ACRONYMS USED

AC	air conditioning
ADS	automatic depressurization system
CIA	containment instrument air
CRDH	control rod drive hydraulic
CRS	control room supervisor
ECCS	emergency core cooling system
EDG	emergency diesel generator
FSAR	Final Safety Analysis Report
HP	health physics
I&C	instrumentation and control
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
LCS	Licensee Controlled Specifications
NRC	U.S. Nuclear Regulatory Commission
PER	problem evaluation request
PPM	Plant Procedures Manual
RHR	residual heat removal
RWP	radiation work permit
SPC	suppression pool cooling
SSC	structures, systems, and components
TS	Technical Specifications
VET	voluntary entry into TSAS/RFO/risk significant activity approval (worksheet)
VIO	violation
WNP-2	Washington Nuclear Project-2