ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Inspectors:	S. A. Boynton, Senior Resident Inspector G. W. Johnston, Senior Project Engineer
Approved By:	H. J. Wong, Chief, Reactor Projects Branch E

Attachment Supplemental Information

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/98-03

Operations

- Inadequate self-checking and peer checking resulted in an operator error that deenergized nonvital Bus SM-2 and started the Division III emergency diesel generator. Operations personnel actions in response to the transient were appropriate and prompt. The licensee's root cause analysis and corrective actions effectively addressed the human performance concerns (Section O1.1).

- One instance was identified in which an operating crew did not demonstrate a conservative approach to equipment operation when a nonvital lighting panel, with an unidentified ground, was reenergized without an understanding of the source of the ground or a troubleshooting plan to identify the source (Section O1.2).

Maintenance

- Poor material condition of the plant service water (TSW) system resulted in a leak that challenged the integrity of the control room envelope as water was able to penetrate through a concrete slab interface in the control room ceiling, a boundary credited by the licensee's flooding analysis. The licensee is currently implementing an improvement plan that should adequately address the material condition deficiencies in theTSW system (Section M2.1).

Engineering

- In reviewing the testing requirements for the standby gas treatment system, the inspector identified the potential for the system floor drains to present a bypass pathway around the filters. In response to the inspector's concerns, the licensee took appropriate action to verify that the current leakage is acceptable, and to develop a long-term monitoring program for this potential unfiltered leakage path (Section E1.1).

- Licensee personnel improperly applied surveillance requirement 3.0.2 to program surveillances in the administrative section of Technical Specifications (TS). As a result, a 25 percent surveillance interval extension was inappropriately utilized for several technical programs (Section E8.2).

- A number of deficiencies were identified in the implementation of the licensee's leakage surveillance and prevention program. Specifically, procedures for performing visual and integrated leakage inspections on the standby gas treatment system, the containment monitoring system (CMS), and the post accident sampling system (PASS), were inadequate in that they failed to identify all of the appropriate system components to be monitored (Section E8.3).

Plant Support

- Licensee corrective actions to address weaknesses in implementing the transient combustible control program have not been effective in addressing the root cause and precluding repeat noncompliances with procedural requirements. The root cause of these noncompliances appeared to be a lack of understanding of fire protection requirements and inattentiveness to fire protection labeling on the part of plant personnel (Section F1.1).

Report Details

Summary of Plant Status

The plant began the inspection period at 100 percent power. On February 3, power was reduced to 70 percent in response to a loss of electrical power to nonvital Bus SM-2. The loss of electrical power resulted in the loss of one of the condensate/condensate booster pump sets. Following equipment restoration, the plant was returned to full power on February 4. On March 1, power was reduced to 60 percent to facilitate investigation and repair of an apparent tube leak in main condenser Waterbox C. Subsequent troubleshooting efforts were unsuccessful in identifying the leaking tube, and power was returned to 100 percent on March 5.

On March 11, a failure of the instrument air line to main steam Line D inboard isolation valve resulted in the inadvertent closure of the valve. The ensuing transient led to the closure of all of the main steam isolation valves and a high flux reactor scram. To facilitate troubleshooting and repair of the broken air line, the licensee placed the plant in Mode 4 on March 12. The plant remained in Mode 4 for the balance of the inspection period. The details of this event will be documented in NRC Inspection Report 50-397/98-05.

I. Operations

O1 Conduct of Operations

O1.1 Inadvertent Deenergization of Bus SM-2, February 3, 1998

a. Inspection Scope (71707)

The inspector reviewed the licensee's Incident Review Board (IRB) report conclusions and the recommended corrective actions of Problem Evaluation Request (PER) 298-0102, following an inadvertent deenergization of Bus SM-2 on February 3, 1998.

b. Observations and Findings

The licensee conducted an IRB investigation of the February 3, 1998, deenergization of Bus SM-2. The result of that IRB review showed that the event was principally due to human performance induced error. The licensee's identified root cause was that the reactor operator responsible for the event failed to "fully internalize the value of Peer Checks and Self-Checking."

The event occurred during performance of PPM 2.7.1.A, "6900 Volt and 4160 Volt AC Electrical Power Distribution System," to transfer Bus SM-1 from Transformer TR-N to Transformer TR-S to support surveillance testing of Diesel Generator DG-1. The actions before commencing the transfer included a prejob brief and a discussion of peer checking. Step 5.16.10 included action to "green flag" the TR-N/SM-1 Switch CB-N1/1. The action to green flag a switch is to ensure the indicator flag on the switch agrees with the lamp indication. In the actions to transfer to TR-S, Breaker CB-S1 is allowed to

open automatically. The switch flag, however, does not automatically reset. Just before the step, the operator responded to an alarm on the adjacent panel. When the operator returned to the activity, he inadvertently opened the TR-N feeder breaker Switch CB-N1/1 to Bus SM-2 rather than "green flagging" the CB-S1 switch. The action to green flag a switch is the same action as manually opening the breaker. This resulted in the loss of Bus SM-2 and subsequent tripping of major loads. The inspector noted that the switches were side by side on the the control panel.

Following the inadvertent loss of Bus SM-2, several major loads in the secondary side of the plant were lost. These included a condensate pump and the associated condensate booster pump. The loss of these two pumps required a prompt reduction of power, by reducing recirculation flow from 100 percent of full power to approximately 65 percent. A low voltage was also sensed on Bus SM-4, the Division III 4160V vital bus normally supplied by SM-2, which resulted in a start of the Division III emergency diesel generator. From a review of the post event chart traces, the inspector determined that the event did not compromise any operating limits. This was due to prompt action by control room operators in reducing reactor power.

In the IRB report, the reviewers noted that the operator responsible for the event had been under some personal stress and was tired at the time of the event. The inspector discussed these aspects with the IRB chairman, observing that the IRB report described these as "noncontributing" causes. The IRB chairman stated that the individual involved raised these points as possible contributors to his performance, and that the IRB discounted them. The IRB interviews of other control room personnel and supervisors found that the operator had not exhibited outward signs of stress or fatigue. The operator had recently returned to shift duties from an extended assignment, which had kept him from normal watch standing duties. The IRB chairman further stated that considerable change in control room conduct of operations had occurred over the period before the operator's return to shift duties. The chairman, further, indicated his view that the operator had not internalized the expectations involved in performing tasks governed by procedure, especially with regard to self-checking and peer checking.

The licensee's immediate corrective actions included a station wide standdown on February 3, 1998, to review this event and other recent personnel errors. An entry was placed in the operations night orders referring to the stand down, reiterating expectations regarding the contributing factors. Shift managers were instructed to evaluate crew members with regard to self-checking. The operations manager stated that the concept of peer checking had been emphasized as a management expectation, but had been left for each shift manager to implement. This had led to variation in the level of expectation and implementation. Therefore, the operations manager directed that management expectations with regard to peer checks be documented in the appropriate Operating Instructions.

A Nuclear Safety Assurance Department (NSAD) quality department surveillance report (298-014) indicated that an NSAD representative observed the recovery from the loss of SM-2. The observer noted the operator logs did not mention that the chemistry

department was notified of the power reduction greater than 15 percent. Followup inquiry by NSAD indicated that the plant laboratory chemistry supervisor believed that operations notification was adequate. No primary chemistry sample was taken, as the noble gas release rate of 919.3 microcuries per second was below the sampling requirement of greater than 15,000 microcuries per second.

The failure to properly perform PPM 2.7.1A, Revision 3, step 5.6.10, to "green flag" the TR-N/SM-1 Switch CB-N1/1 was identified as a violation of Technical Specification (TS) 5.4.1.a, which requires, in part, procedures to be developed, implemented and maintained for each of the surveillances directed by plant TS. This nonrepetitive, self-revealing and corrected violation is being treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-397/98003-01). Licensee Event Report 50-397/98001-00, relating to the same event, is also closed.

c. Conclusions

Inadequate self-checking and peer checking resulted in an operator error that deenergized nonvital Bus SM-2 and started the Division III emergency diesel generator. Operations personnel actions in response to the transient were appropriate and prompt. The licensee's root cause analysis and corrective actions effectively addressed the human performance concerns.

O1.2 Control Room Logs

a. Inspection Scope (71707, 62707)

The inspector reviewed the control room log and problem evaluation requests (PERs) on a daily basis to evaluate the accuracy of the logs, and the actions taken by the licensee in response to conditions identified.

b. Observations and Findings

Over the past several months the inspector identified several discrepancies in the control room logs that were generally administrative in nature (for example, wrong individual identified as shift manager, wrong shift manager verifying log entries for a shift). Although the errors were administrative, the errors made it difficult to determine whether of one of the shift managers had maintained appropriate watch standing qualifications. The inspector noted that the shift manager had not been routinely assigned to the control room during the fourth quarter of 1997. The licensee reviewed payroll records and interviewed personnel to verify the requirements of 10 CFR Part 55 had been satisfied. The technical accuracy of the logs was noted to be very good.

On March 1, a control log entry was made for reenergizing a lighting panel in the turbine building to support work in the condenser bay. This entry was noteworthy in that the panel in question had an existing work request against it due to an unidentified electrical ground. Upon reenergizing the panel a 4-5 amp ground was noted on the supply

bus (MC-2D-A) to the lighting panel. Subsequent field walkdowns found the neutral ground resistor for Transformer 2-21 to be very hot.

This inspector discussed this entry with the responsible shift manager and found that no troubleshooting activities had been planned or performed to identify the ground prior to reenergizing the panel. The shift manager, in clearing the danger tag associated with the supply breaker, made the assumption that the ground was due to one of the mercury vapor lamps powered from the panel, based upon experience. That experience had shown historically that if the ground was in a lamp, the ground would be eliminated when the lamp cooled down. The inspector disagreed with this approach in that the decision to reenergize the lighting panel was based upon an assumption and not on actual data for the existing condition. The result was a potential fire hazard and the potential for damage to electrical equipment. The impacted equipment was not safety-related.

On March 8, control room operators identified what appeared to be an unexpected oscillation transient on reactor feedwater Turbine B. The shift technical advisor (STA) had also been touring reactor feedwater pump Room B at the time of the event. According to the PER initiated for the event, the STA repeated his actions in the pump room in an attempt to repeat the transient to see if he may have caused or contributed to the event. Further review by the licensee determined that the actions taken by the STA were actually benign and that the PER did not accurately reflect those actions. The inspector also noted that although the operating crew determined the event warranted the initiation of a PER, no control room log entry was made to note it.

c. Conclusions

The technical accuracy of the control room logs was found to be generally very good, with the exception of the omission of a log entry for a feedwater turbine transient and a few minor administrative errors.

One instance was identified in which an operating crew did not demonstrate a conservative approach to equipment operation when a nonvital lighting panel, with an unidentified ground, was reenergized without an understanding of the source of the ground or a troubleshooting plan to identify the source.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns (71707)

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

- Reactor Core Isolation Cooling
- Standby Gas Treatment, Trains A and B

- Residual Heat Removal, Trains A, B, and C
- High Pressure Core Spray

Each of the systems was found to be properly aligned for the current operating mode. No deficiencies were identified by the inspectors that had not already been identified by the licensee. A review of plant logs and operating data indicated that system reliability and availability remain well above maintenance rule performance guidelines.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

a. Inspection Scope (62707, 61726)

The inspectors observed the following work activities:

• OSP-HPCS/IST-Q701	High Pressure Core Spray (HPCS) System Operability Test
• WO# KRW0	Troubleshoot and Repair Instrument Air Line to MS-V-22D
• PPM 6.2.4	New Fuel Inspection

b. Observations and Findings

Each of the work activities was adequately implemented. Radiological protection practices during new fuel receipt inspections and the drywell maintenance on Valve MS-V-22D were appropriate. The ALARA planning and execution for the work in the drywell was effective in minimizing the dose to workers. Foreign material controls were also properly implemented for these activities.

Operations performance during the high pressure core spray (HPCS) pump operability test was adequate. Proper peer checking was observed for control board manipulations. Three-way communications were not consistently utilized, contrary to management expectations. The control room supervisor (CRS) authorized the stroking of manual Valve HPCS-V-6 (HPCS keep-fill pump discharge check valve) out of sequence from the procedure. This change to the order of the procedure was contrary to the guidance in SWP-PRO-01, "Description and Use of Procedures and Instructions," which states that procedure steps within TS surveillances should be performed in the sequence specified unless otherwise designated by the procedure.
Procedure OSP-PCS/IST-Q701 did not specifically authorize this practice. The system engineer that was monitoring the surveillance, noted the discrepancy and informed the

operators that the manual valve had to be stroked with the keep-fill pump operating. The procedure steps were then repeated satisfactorily in the order specified.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Material Condition Deficiencies in TSW

a. Inspection Scope (37551)

The inspector assessed the licensee's actions to resolve a leak that occurred in the service water system on the 525-foot elevation of the Radwaste Building. The inspector also reviewed the licensee's assessment of the leakage in PER 298-0157.

b. Observations and Findings

On February 20, 1998, electricians were installing electrical conduits in an Instrumentation and Controls shop, Room C510, on the 525-foot elevation of the Radwaste Building. During the work, an electrician leaned on a 3/4-inch drain line from a strainer on a TSW line to a room air conditioning Unit WRA-AC-52. The force applied broke the pipe nipple, initiating a leak onto the floor. Water was unable to drain into a local scupper because of an elevated lip. Subsequently, the water flowed out under the door into adjacent spaces and then flowed into two floor drains. The electricians initiated action to isolate the leak and informed the control room operators immediately. Later, the control room operators noted the sound of dripping water in the ceiling overhead. An inspection by the system engineer noted that a small amount of water had gone into the ceiling overhead space and had dripped onto the upper surface of the ceiling tiles. The engineer noted that the leakage was coming from seams in the floor slab from the construction joints in the concrete.

The inspector toured the affected area with the system engineer. From the tour of the spaces, the inspector noted that the floor slab appeared to have been poured during construction in six separate pours. The boundaries between each slab were not cleanly dressed or troweled. The inspector observed a fine crack between the slabs that appeared sufficient for water to penetrate and migrate. This was the apparent cause of the leakage into the ceiling space above the control room.

A barrier impairment was issued for the floor surface for flooding protection because the floor is a credited flooding boundary in the fire protection program. A work request (WR 98000899) was initiated to apply a surface coat to the floor to provide an impervious flooding boundary. Immediate action included the staging of plastic sheeting to provide protection to control panels in the control room. It is unlikely, due to the narrow nature of the cracks in the 525-foot floor slab, that substantial amounts of water could flow into the control room ceiling overhead. A similar configuration was also identified for the 484-foot elevation of the Radwaste Building (cable spreading room), resulting in the issuance of a barrier impairment for that elevation. These actions appeared appropriate in addressing the immediate concerns from possible flooding of

the 525-foot elevation. The licensee initiated action to review other possible locations in other buildings that may be similar to the concerns identified in the Radwaste Building.

The design requirement for flooding protection is in Tech Memo 2103, which was generated for fire barrier analysis, and addresses penetration seals. The design requirement is not identified in the Facility Safety Analysis Report. Currently the floor is required to maintain a 4-inch maximum standing water depth. The leakage through joint cracks in the floor slab does not meet the requirement in the Tech Memo because of the observed leakage. The control room envelope integrity has been demonstrated by surveillance testing and therefore meets the operability requirement in the TS. However, because of the leakage into the control room, the envelope was considered degraded due to the barrier impairment issued for the TSW leakage.

The system engineer for the TSW system indicated that a piping replacement program is in place and a schedule has been developed identifying the portions of piping that are to be replaced during each refueling outage. Prioritization of the replacement was from nondestructive examination and system importance. The engineer stated the licensee's goal is eventually to replace all affected piping. The piping replacement scheduled for Refueling Outage R-13 is small bore supply and return lines to the main steam tunnel coolers (RRA-CC-8 and 9). The reason for this replacement is that a leak in the main steam tunnel could result in a plant shutdown. Other actions to address the material condition of the TSW piping include a silt/rust inhibitor system that has been added to the system and plans to install an anti-biofouling injection system. The silt/rust inhibitor system is currently undergoing testing and troubleshooting.

c. Conclusions

Poor material condition of theTSW system resulted in a leak that challenged the integrity of the control room envelope as water was able to penetrate through a concrete slab interface in the control room ceiling, a boundary credited by the licensee's flooding analysis. The licensee is currently implementing an improvement plan that should adequately address the material condition deficiencies in the TWS system.

III. Engineering

E1 Conduct of Engineering

E1.1 Testing of the Standby Gas Treatment (SGT) System

a. Inspection Scope (37551)

In reviewing the licensee's implementation of its leakage surveillance and prevention program for the SGT system (see Section E8.3), the inspector reviewed the testing performed on the system to ensure it meets its design requirements.

b. Observations and Findings

Licensee TS 3.6.4.3 requires the SGT system to be tested in accordance with the Ventilation Filter Testing Program as described in TS 5.5.7. TS 5.5.7 requires the high efficiency particulate air (HEPA) and charcoal filters of the SGT system to be tested in accordance with ASME-N510-1989, "Testing of Nuclear Air Treatment Systems," to demonstrate that penetration and system bypass is less than 0.05 percent.

The inspector reviewed the 1975, 1980, and 1989 versions of ASME-N510 and found that each of the standards describes the methodology for testing the penetration and bypass of the HEPA and charcoal filters. In the 1975 and 1980 versions of the standard an appendix was included regarding the importance of in-place leak tests. In these versions it was stated that "an installed system can be assumed to have an efficiency equivalent to that of the [factory tested charcoal] sample only if: . . .there are no leaks or bypasses in either the individual [charcoal] cells (factory tests) or the installed system (field tests)." The 1989 version went further and included a specific testing method and frequency for system bypass leakage. The purpose of the test was described as follows:

> Systems using HEPA filters and adsorber banks may contain bypass dampers, ducts, conduits, floor drains, pipe penetrations, etc., which could potentially defeat the purpose of high efficiency nuclear air treatment components. Therefore, it is necessary to perform tests which challenge all these potential bypass leakage paths...

ASME N510-1989 recommends a frequency of once per operating cycle for performing the system bypass leakage test.

The inspector noted that the SGT filter train contains a number of floor drains for directing the flow of fire protection deluge water to the reactor building floor drain system. These drains are normally closed by means of a passive check valve in each drain line. Several of the drains are located in the exhaust section of the filter train, and as such, present a potential bypass leakage path if the check valve failed or was not properly seated. In questioning the licensee on testing of these valves, it was determined that the licensee did not have a program for visually inspecting the valves or a procedure to measure system bypass leakage from the floor drain system.

In response to the inspector's concerns, the licensee attempted to measure the air flow into the common header of the SGT floor drain system to evaluate if any of the check valves may be leaking by their seat. Measurements made with a pitot tube velocimeter at the outlet of the common drain header found that the air velocity into the header on both trains was less than that detectable with the instrument (25 ft/min). The licensee then estimated that leakage flow into the system, via the check valves, was less than 0.5 scfm. The leakage was well within the analysis assumption of 14 scfm.

Although the leakage rate past the check valves was found to be acceptable, the inspector questioned the licensee on the need to test this leakage path on a periodic basis as recommended by ASME N510-1989. The licensee reviewed the testing requirements of the SGT system and did not identify any licensing basis for testing the bypass flow path. Specifically, in implementing the testing requirements of ASME N510-1989, through TS 5.5.7, only testing of the in-place HEPA filters and charcoal adsorbers is required. However, due to the potential consequences of bypass leakage, the licensee agreed that testing would be prudent. As a result, the licensee will be implementing a periodic testing program for the SGT floor drains.

c. Conclusions

In reviewing the testing requirements for the SGT system, the inspector identified the potential for the system floor drains to present a bypass pathway around the filters. In response to the inspector's concerns, the licensee took appropriate action to verify that the current leakage is acceptable, and to develop a long-term monitoring program for this potential unfiltered leakage path.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-397/97018-08: lack of written safety evaluation for removal of information from the fire hazards analysis (FHA) in the Final Safety Analysis Report (FSAR). A more detailed review of the changes made to the FSAR found that the information removed from the FSAR was duplicated in the licensee's combustible loading calculation. In removing that information, the licensee also incorporated the combustible loading calculation into the FSAR by specific reference. Thus, changes to the calculation would then be required to be evaluated under the requirements of 10 CFR 50.59. The inspector found that no changes were made to the calculation in conjunction with the removal of the information from the FSAR. Therefore, the removal of the information, in and of itself, did not constitute a change to the facility and did not require a written safety evaluation.

In response to a number of outstanding calculation modification reports, and several concerns raised by the inspector, as documented in NRC Inspection Report 50-397/97-18, the licensee recently revised the combustible loading calculation. The inspector verified that a written safety evaluation was performed to evaluate those changes and the associated changes to the FHA section of the FSAR.

E8.2 (Closed) Unresolved Item 50-397/97018-04: inappropriate application of TS 4.0.2 to allow for a 25 percent extension to certain surveillance intervals. This item was opened based upon the licensee's use of TS 4.0.2 when scheduling the integrated leakage surveillances required by TS 6.8.4.a.2 for evaluating primary coolant leakage outside of containment. In resolving this issue, the licensee reviewed each of the 72 integrated leakage surveillances performed, in accordance with TS 6.8.4.a.2, since initial operation. The licensee found that the use of TS 4.0.2 resulted in 27 instances where the 18-month surveillance interval was exceeded. The licensee also found that TS 4.0.2 was applied

to TS 6.8.4.d.5. TS 6.8.4.d.5 (Improved Technical Specification (ITS) 5.5.4.e) requires every 31 days a determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and calendar year. Although the application of Surveillance Requirement (SR) 3.0.2 (old TS 4.0.2) is not explicitly referenced, a 25 percent extension has been routinely applied to this SR.

As noted in NRC Inspection Report 50-397/97-18, the licensee's implementation of ITS now allows for the use of a 25 percent surveillance interval extension (SR 3.0.2) for those surveillances required by TS 5.5.2 (equivalent to old TS 6.8.4.a).

The licensee determined that the root cause of the misapplication of TS 4.0.2 (SR 3.0.2) was personnel error in that there was a general misunderstanding of how TS 4.0.2 (SR 3.0.2) applied to SRs in the Administrative Controls section. Specifically, it was not understood that application of TS 4.0.2 (SR 3.0.2) to surveillances under administrative TS was not provided unless explicitly referenced. In response, the licensee developed a set of rules for determining the applicability of SR 3.0.2 to SRs that are outside of Sections 3.1 through 3.10 of ITS. The inspector reviewed the licensee's methodology and found that, if properly implemented, the rules would result in the proper application of SR 3.0.2. The licensee has established a corrective action to train licensing personnel and program managers on this methodology.

To meet the 31-day interval requirement of TS 5.5.4.e, the licensee is determining the cumulative and projected doses from radioactive effluents twice each month until a TS amendment is approved to allow the application of SR 3.0.2.

The failure to perform the integrated leakage surveillances on an 18-month interval was identified as a violation of TS 6.8.4.a.2 (VIO 50-397/98003-02). The failure to determine cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and calendar year every 31 days was identified as a violation of TS 5.5.4.e (VIO 50-397/98003-03).

E8.3 (Closed) Unresolved Item 50-397/97018-03: undefined process for evaluating and correcting leakage from the SGT system in accordance with TS 5.5.2. TS 5.5.2 requires that biennial leakage inspections and periodic visual inspections be performed on those systems outside of containment that would contain highly radioactive fluids following a postulated loss-of-coolant accident (LOCA). SGT is included as one of the systems to monitor under TS 5.5.2. However, PPM 1.5.6, "Leakage Surveillance and Prevention Program," Revision 8, the licensee's procedure that implements the TS 5.5.2 program, did not define those components of SGT that need to be monitored for leakage and did not provide for periodic visual inspections.

Subsequent to the findings noted in NRC Inspection Report 50-397/97-18, the licensee initiated a broad review of its implementation of the requirements of TS 5.5.2. This review included the basis for which SGT is included as a system under TS 5.5.2, and detailed reviews of the procedures that implement the inspection requirements, as referenced by PPM 1.5.6.

SGT System: Subsequent evaluations by the licensee have been unsuccessful in determining the technical basis for the inclusion of the SGT system in TS 5.5.2. In fact, the licensee's original commitment to implementing the leakage surveillance and prevention program, as described in the FSAR, did not list the SGT system as one of the systems to be monitored.

In reviewing the operating characteristics of the system, the inspector found that the system isolates from containment upon the initiation of a LOCA. The leakage from the containment isolation valves is monitored through the licensee's inservice testing program. The only portion of the system downstream of the containment isolation valves in which leakage would be of concern appeared to be that section of the system between the outlet of the charcoal filters and the inlet of the fans. Air in-leakage in this portion of the system would be drawn in to the fans and would bypass the filter unit. The licensee's analysis for evaluating offsite doses following a postulated LOCA assumes that 14 scfm of unfiltered air is discharged by SGT. This takes into account leakage from joints, valve packing, and the fans' shaft seals. The testing performed following initial installation of the system verified that the 14 scfm was a conservative assumption.

The licensee plans to address visual inspections of the SGT system through a revision to the routine operability surveillance procedures.

CMS: Concerns were also identified with the licensee's leakage monitoring of the CMS. Prior to the licensee's implementation of ITS, PPM 7.4.6.3.4.2, "Excess Flow Check Valve Test of Containment Atmosphere and Suppression Pool Level," was being credited for the biennial integrated leakage surveillance required by old TS 6.8.4.a.2 (ITS 5.5.2). However, PPM 7.4.6.3.4.2 only verified operability of the CMS excess flow check valves and checked for leakage at the containment penetration isolation and test valves. No leakage inspections were required for components downstream of the excess flow check valve. Additionally, the inspector noted that PPM 7.4.6.3.4.2 was canceled upon implementation of ITS and that a new procedure has not yet been approved.

A new procedure to test the CMS excess flow check valves was in the review process at the conclusion of the inspection. The system engineer was also working with the TS 5.5.2 program manager to determine what additional components would need to be inspected as part of the biennial leakage surveillance.

PASS: Similar to the SGT system, a procedure had not been identified for performing periodic visual inspections of the PASS system. The licensee is in the process of determining an appropriate mechanism and frequency for performing these inspections. The licensee's biennial integrated leakage inspection is proceduralized in PPM TSP-PASS-B801, Revision 0, "Post Accident Sampling Leakage Surveillance." The system engineer's review of this procedure identified a number of components within the PASS system that are not evaluated during the biennial surveillance. The components included, but were not limited to, the isolation valves from the residual heat removal

system and sampling valves within the PASS sample rack. A revision to PPM TSP-PASS-B801 has been initiated to incorporate the findings of the system engineer.

Conclusions: The existing monitoring requirements for the other systems covered under TS 5.5.2 were found to be generally adequate. However, the failure of PPM 1.5.6 to adequately address the SRs of TS 5.5.2 for the SGT, CMS, and PASS systems was identified as a violation of TS 5.4.1.e. (VIO 50-397/98003-04).

IV. Plant Support

F1 Conduct of Fire Protection Activities

F1.1 Inadequate Corrective Actions to Address Failures to Control Transient Combustibles

a. Inspection Scope (71707, 71750)

The inspector reviewed the licensee's actions to address several examples of inadequate control of transient combustibles in the reactor building, as described in NRC Inspection Report 50-397/97-18. Several walkdowns of vital fire protection areas were also performed to evaluate the adequacy of those actions.

b. Observations and Findings

NRC Inspection Report 50-397/97-18 identified several examples where the requirements of PPM 1.3.10C, "Control of Transient Combustibles," were not followed for transient combustible materials left in high radiation areas of the reactor building. In response to the identified concerns, the licensee performed a root cause analysis and has implemented a corrective action plan through the associated PER. The plan included the issuance of an e-mail message from the plant general manager to department managers and a discussion between the operations manager and the operations shift crews of the fire protection concerns. Additional discussions between the other department managers and their staffs are planned, but have not yet been completed. The licensee's root cause analysis identified weaknesses in plant staff knowledge of the requirements of PPM 1.3.10C. Specifically, it was concluded that plant staff may not have a clear understanding of what constitutes combustible material that would require controls under PPM 1.3.10C. However, the corrective action plan only called for a review of the root cause and corrective actions for the associated PER in update training sessions. No short-term actions were taken to focus the plant staff's attention on the transient combustible permit process and the treatment of combustible-free zones.

On February 18, the inspector identified unattended transient combustibles in a safety-related instrument rack room on the 501-foot elevation of the reactor building. The combustible materials were introduced into the room as a part of planned maintenance activities that also established a contamination zone and step-off pad in

the room. The instrument rack room is designated as a combustible-free zone in accordance with PPM 1.3.10C. As such, unattended transient combustibles are not allowed in this room. The licensee subsequently initiated PER 298-0144 to document the procedure noncompliance. The combustibles were removed from the room and a transient combustible permit was issued by the plant fire marshal to allow for small amounts of transient combustibles in the room with the requirement that they be constantly attended.

On March 6, the inspector again identified unattended combustibles in the same instrument rack room. The transient combustibles (small plastic containers, tygon tubing, and electrical multimeter) were apparently staged for an instrumentation and controls surveillance. The inspector attended the material until the shift support supervisor (SSS) arrived. The SSS subsequently left the area without removing the material or leaving another individual in attendance. Although the inspector noted to the CRS that the material found did not comply with the requirements of PPM 1.3.10C, a PER was not initiated until the inspector discussed this concern with the fire protection engineer on March 10.

The inspector noted that a sign was posted in the interior of the instrument rack room indicating that the room is a combustible-free zone. The inspector also noted that the transient combustible permit, issued following the concerns identified on February 18, was posted in the stairwell adjacent to the room, but not readily visible to individuals accessing the room. Both of the two occurrences identified by the inspector indicate that plant personnel were not being attentive to fire protection program labeling and did not understand the requirements of PPM 1.3.10C for what constitutes transient combustible material. The failure of the SSS to constantly attend or remove the material from the instrument rack room on March 6, and the failure of the CRS to initiate a PER for the procedure noncompliance, both underscore the concern. They also indicate that the licensee's actions to address personnel knowledge deficiencies in implementing the transient combustible control program, in response to the concerns identified in NRC Inspection Report 97-18, were inadequate to preclude repeat procedure noncompliances. The failure to take prompt and adequate corrective actions for improper control of transient combustibles in the reactor building, a condition adverse to quality, was identified as a violation of 10 CFR Part 50, Appendix B, Criterion XVI (VIO 50-397/98003-05).

c. Conclusions

Licensee corrective actions to address weaknesses in implementing the transient combustible control program have not been effective in addressing the root cause and precluding repeat noncompliances with procedural requirements. The root cause of these noncompliances appeared to be a lack of understanding of fire protection requirements and inattentiveness to fire protection labeling on the part of plant personnel.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management after the conclusion of the inspection on March 25, 1998. 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.

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

50-397/98003-01	NCV	failure to properly perform PPM 2.7.1A
50-397/98003-02	VIO	failure to perform the integrated leakage surveillances on an 18-month interval
50-397/98003-03	VIO	failure to determine cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and calendar year every 31 days
50-397/98003-04	VIO	failure of PPM 1.5.6 to adequately address the SRs of TS 5.5.2 for the SGT, CMS, and PASS systems

50-497/98004-05	VIO	failure to take adequate corrective actions for improper control of transient combustibles in the reactor building

Closed

50-397/98001-00	LER	voluntary report of automatic start of HPCS DG due to operator error
50-397/98003-01	NCV	failure to properly perform PPM 2.7.1A
50-397/97018-03	URI	undefined process for evaluating and correcting leakage from the SGT system in accordance with TS 5.5.2.
50-397/97018-04	URI	inappropriate application of TS 4.0.2 to allow for a 25 percent extension to certain surveillance intervals
50-397/97018-08	URI	lack of written safety evaluation for removal of information from the FHA in the FSAR

LIST OF ACRONYMS USED

CMS	containment monitoring system
CRS	control room supervisor
FSAR	Final Safety Analysis Report
HEPA	high efficiency particulate air
HPCS	high pressure core spray
IRB	incident review board
LER	licensee event report
ITS	Improved Technical Specifications
LOCA	loss-of-coolant accident
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
NSAD	Nuclear Safety Assurance Department
PASS	post accident sampling system
PER	problem evaluation request
PPM	Plant Procedures Manual
SR	surveillance requirement
SGT	standby gas treatment
SSS	shift support supervisor
TS	Technical Specifications
TSW	plant service water
URI	unresolved item
VIO	violation
WNP-2	Washington Nuclear Project-2