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 RECIP. NAME RECIPIENT AFFILIATION
 Document Control Branch (Document Control Desk)

SUBJECT: Responds to NRC ltr re violations noted in insp repts
 50-397/93-18 & 50-397/93-24. Corrective actions: prepared
 substitution evaluation 1786 & counseled involved personnel.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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September 8, 1993
G02-93-228

Docket No. 50-397

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: WNP-2, OPERATING LICENSE NO. NPF-21
 NRC INSPECTION REPORTS 93-18 AND 93-24
 REPLY TO APPARENT VIOLATIONS

The Nuclear Regulatory Commission (NRC) conducted inspection activities at Washington Nuclear Plant No. 2 (WNP-2) in the period May to August of this year. These inspections are described in NRC Inspection Report Nos. 50-397/93-18 and 50-397/93-24. The reports identify 13 examples of apparent violations involving "failures to follow procedures."

The NRC also recently completed another related inspection which is described in NRC Inspection Report No. 50-397/93-29. The apparent violations described in Report 93-29 as well as those in Reports 93-18 and 93-24 are being considered for escalated enforcement action. An enforcement conference has been scheduled for September 22, 1993.

The Supply System recognizes that there are procedural adequacy and compliance issues raised in each report and, thus, combining the discussion of the issues for the enforcement conference is logical. However, given the low safety significance of the individual matters in Report Nos. 93-18 and 93-24, and the Staff's focus on the programmatic implications of procedural non-compliance, we believe that the enforcement conference will be more effective if the details of the 93-18 and 93-24 issues have already been fully described. In this regard, the Supply System provides a discussion of the 93-18 and 93-24 issues in Attachments 1 and 2.

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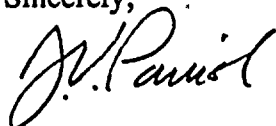
NRC INSPECTION REPORTS 93-18 AND 93-24; REPLY TO APPARENT VIOLATIONS

We believe that this approach is in the best interest of both the NRC and the Supply System, and that it will allow both parties to have a more focused and meaningful discussion on the issues of broader concern. Consistent with your request in the transmittal letter to Report 93-29, we plan to discuss the following issues: (1) the broad implications of procedural non-compliance; (2) the specific issues raised in Report 93-29; and (3) management oversight as it is related to these matters. In addition, while the Attachments describe specific corrective actions for each apparent violation from Reports 93-18 and 93-24, the Supply System also plans to discuss broader corrective actions for these matters as a whole.

Since we do not plan to specifically address the details of Report Nos. 93-18 and 93-24, we request that any questions concerning these matters be brought to our attention as soon as possible before the enforcement conference. Of course, if questions remain outstanding, we will address them as appropriate during the conference.

Should you have any questions regarding this submittal, please contact Douglas W. Coleman at (509) 377-4342.

Sincerely,



J. V. Parrish (Mail Drop 1023)
Assistant Managing Director, Operations

DWC/PDR/bk

Attachments: 1) Response to Apparent Violations Cited in NRC Inspection Report 93-18
2) Response to Apparent Violations Cited in NRC Inspection Report 93-24

cc: BH Faulkenberry - NRC RV
NS Reynolds - Winston & Strawn
JW Clifford - NRC
DL Williams - BPA/399
NRC Site Inspector - 901A

ATTACHMENT 1

RESPONSE TO APPARENT VIOLATIONS

CITED IN NRC INSPECTION REPORT 93-18

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NRC INSPECTION REPORT 93-18: RESPONSE TO APPARENT VIOLATIONS

A. SUMMARY: APPARENT VIOLATION 93-18-01

Stainless steel accumulator (PIN 112D3405G001) was approved for replacement of the Control Rod Drive (CRD) Hydraulic Control Unit (HCU) accumulators by Procurement Evaluation 3395, Rev. 0. This change represented a substitution since the original accumulator (PIN 921D595G001) was made of carbon steel, had a smaller diameter, and had different temperature and pressure ratings. Procurement Evaluation 3395, Rev. 0, referenced Basic Design Change (BDC) 55-0005-0A for the substitution. However, this BDC was the design change document for the substitution of a carbon steel accumulator that was similar to the original and essentially only differed in part number.

Thus, the BDC was erroneously referenced by the Procurement Evaluation since it was prepared for a different accumulator. However, due to the referencing error, a Substitution Evaluation was not performed for the stainless steel accumulator prior to its use. The failure to perform a Substitution Evaluation is contrary to Plant Procedures Manual (PPM) 1.17.2, "Procurement Engineering Reviews."

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation.

ROOT CAUSE

In 1986 WNP-2 pursued obtaining replacement parts for the carbon steel Control Rod Drive (CRD) Hydraulic Control Unit (HCU) accumulators from the "to be terminated" Tennessee Valley Authority (TVA) Phipps Bend Nuclear Plant. Since the part number of the Phipps Bend carbon steel accumulators was different than the WNP-2 part number, Procurement Evaluation 348 was issued on December 24, 1986, and subsequently revised by Substitution Evaluation 10 to authorize procurement under the Phipps Bend part number.

On November 15, 1989, Material Problem Report MP 100111 was initiated to document that the Phipps Bend CRD HCU accumulator had been declared "obsolete" by General Electric (GE). It had been superseded by a stainless steel accumulator that was carbon steel lined with a chromium steel overlay. To accommodate this change, Procurement Evaluation 3395, Revision 0, was prepared on December 11, 1990, to supersede and void Procurement Evaluation 348, and authorize procurement of the new stainless steel substitute accumulators. During the preparation of this evaluation, the design change (BDC 55-0005-0A) that allowed the substitution of the Phipps Bend accumulators for the original WNP-2 accumulators was inappropriately referenced and was not discovered during the checking or approval reviews. This resulted in documentation that allowed

substitution of the stainless steel accumulators for the Phipps Bend accumulators using BDC 55-0005-0A, with no additional design change requirements. Purchase Order P30591-053 was subsequently issued on July 23, 1992, to procure the stainless steel accumulators on this basis.

The root cause for Apparent Violation 93-18-01 was a personnel work practices error. The Supply System nonlicensed engineer that prepared Procurement Evaluation 3395, Revision 0, failed to recognize that BDC 55-0005-0A was not adequate to allow substitution of the stainless steel accumulators.

SIGNIFICANCE & MITIGATING FACTORS

The substitute accumulators were supplied by General Electric Co., the original equipment manufacturer (OEM), who is an approved Quality Class (QC) 1 supplier. The subsequent Substitution Evaluation (1786), approved on June 10, 1993, showed that the new stainless steel accumulators were satisfactory for substitution and, therefore, there was no operability concern. The Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

The following corrective actions have been implemented:

1. Substitution Evaluation 1786 was prepared and approved for substitution of the stainless steel accumulators.
2. A review of a representative sample of past work by the individual responsible for this apparent violation was performed. The results of the review indicate that this error was an isolated case.
3. The personnel involved with this apparent violation were counseled.

B. SUMMARY: APPARENT VIOLATION 93-18-03

On June 15, 1993, an NRC inspector reviewed the Locked Valve Checklist (checklist) for valves in the drywell to determine whether they were in their proper position and locked in accordance with PPM 1.3.29, "Locked Valve Checklist," and the WNP-2 Final Safety Analysis Report (FSAR). The inspector determined that the handwheels for valves RRC-V-51A and RRC-V-51B were removed and secured to a "nearby seismic support." PPM 1.3.29 requires "[a]ll locked valves greater than 2 inches shall be fixed in place using a shackle lock and chain." "The inspector concluded that the licensee had not met the requirements of PPM 1.3.29, because the method used to lock the above listed valves was not adequate to fix the valves in place."

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation.

ROOT CAUSE

In previous revisions to PPM 1.3.29, the following statement was contained in the Comments section: "Handle locked to structural supports." Operators understood this to be an optional means of securing the handwheels (i.e., instead of to the valve body itself). However, PPM 1.3.29, Rev. 17, did not include the comment.

Because the operators were aware that removing handles and locking to structural supports had been an acceptable practice before, they did not consult the current revision of the procedure for guidance. Therefore, the root cause is considered to be inattention to detail in implementing PPM 1.3.29.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

Prior to the NRC's inspection, the Supply System determined that Reactor Recirculation (RRC) Drain Valves RRC-V-51A, RRC-V-51B, RRC-V-52A, and RRC-V-52B were not required to be locked closed and, in fact, should be removed from the PPM 1.3.29 checklist. These valves are contained in drain lines that are uncapped and hard-piped to the Containment Drywell Equipment Drain Sump, and unauthorized opening of the valves or leakage past them would be detected by flow and sump level alarms.

Contrary to the Inspection Report, inadvertently opening the valves is not expected to result in the reactor water level being "lowered to 2/3 core height." A decrease in reactor water level would result in an increase in the Containment Drywell Equipment Drain Sump to Reactor Building Equipment Drain Sump gravity drain flow. Assuming no mitigating operator or automatic actions are taken for the decreasing reactor water level, the increased Drywell sump to Reactor Building sump flow would be monitored and recorded on the associated Leak Detection (LD) System control room flow recorder, and annunciated by the high flow alarm. In addition, the rising Reactor Building sump water level would be detected by the sump high water level alarm. Appropriate Operator response actions would be taken for the unidentified Containment Drywell leakage. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

The following corrective actions have been implemented:

1. The removal of RRC-V-51A, RRC-V-51B, RRC-V-52A, and RRC-V-52B from PPM 1.3.29, Revision 18, was approved on July 6, 1993.



2. A review of valves locked in accordance with PPM 1.3.29 was completed on August 6, 1993, to determine if there were any other valves that had the handwheels removed. No other valves listed in the procedure were found with the handwheels removed.
3. Operations personnel will be cautioned to question any valve that cannot be signed off exactly as the valve checklists indicate. This action will be completed by October 1, 1993.

C. SUMMARY: APPARENT VIOLATION 93-18-04

On May 27, 1993, an NRC Inspector found the Division 1 Battery Charger Room Fire Door (C-221) propped open with a block of wood. No licensee personnel were in the area guarding the door, and a fire impairment checklist was not present for the open door as required by PPM 1.3.10, "Fire Protection Program."

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation.

ROOT CAUSE

Investigation into this matter indicated that there were two previous Fire Protection System Impairment Checklists (92-280 and 93-078) against Fire Door C-221 for a latch problem and a hole in the door. Since these were not planned impairments, the procedural requirement for posting the Impairment Checklist (permit) on the affected fire door is not clearly specified. However, in practice, the permits have been routinely posted on or adjacent to the affected door. Members of the Battery Test Discharge Crew verified that permits had been posted at Fire Door C-221, but not for an open door. As a result of the existing impairments, the fire door was placed on hourly fire tours as compensatory action.

As required by PPM 1.3.10, a member of the Battery Test Discharge Crew went to the Radwaste Control Room to initiate an "impairment form" for a group of fire doors to facilitate battery charger maintenance. The Shift Support Supervisor (SSS) reviewed existing impairments of the affected fire doors and found that there were existing fire impairments on all the affected doors, including Fire Door C-221. The SSS believed that an additional permit was unnecessary because PPM 1.3.10 did not contain a requirement for "cascading" impairments. In addition, there were ongoing fire tours of the doors and the duration of the battery charger maintenance was relatively short. This was discussed with a member of the Battery Test Discharge Crew.

The SSS subsequently gave the Battery Test Discharge Crew permission to prop open the affected fire doors as necessary, including Fire Door C-221, without initiating an additional permit. The Battery Test Discharge Crew members were under the impression that since the fire doors were already impaired, with compensatory action in progress, no additional controls were necessary. Later discussions with the Fire Marshall concluded that this logic was correct and that PPM 1.3.10 was not clear in this area. The root cause for this apparent violation was that PPM 1.3.10 was less than adequate in that the procedure does not address the condition of "cascading" impairments.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

During the period Fire Door C-221 was propped open, fire tours were in effect for the door as compensatory action in accordance with PPM 1.3.10. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

The following corrective actions were implemented:

1. The Plant Fire Marshall issued an IOM on June 1, 1993, to Maintenance Management to remind maintenance personnel of the impairment requirements of PPM 1.3.10. The procedure sections relevant to this matter were discussed with maintenance personnel during their daily staff/safety meeting.
2. PPM 1.3.10 will be extensively revised by December 31, 1993. This revision will address "cascading" impairments and requirements for posting the Impairment Checklist for unplanned impairments.

D. SUMMARY: APPARENT VIOLATION 93-18-05

Example 1

On May 19, 1993, an NRC Inspector informed the Supply System that the sign/barrier posting a radiation area located on the 548-foot elevation of the Reactor Building " ... was hanging from a single padeye...." Since the subject area contained radiation levels in excess of 2.5 mrem/hr, PPM 11.2.7.1, "Area Posting," requires that the area be conspicuously posted as a radiation area.

Example 2

At 0330 hours on May 27, 1993, an NRC Inspector informed the Supply System that the radiation barrier at the entrance to the Hydrogen Recombiner room on the 441-foot elevation of the Turbine Building was removed by a Supply System employee during passage, but not restored. Since the subject area contained radiation levels in excess of 2.5 mrem/hr, PPM 11.2.7.1 requires that the area be conspicuously posted as a radiation area.

Example 3

On May 28, 1993, an NRC Inspector identified that the area behind Sample Rack (SR) PO26, located on the 522-foot elevation of the Reactor Building, was not properly posted as a high radiation area (HRA). The area behind the rack was not a HRA, but it could allow access to an HRA. The potential pathway was not typically utilized because the clearance between the rack and the containment wall was only about 16 inches. Nevertheless, the NRC Inspector observed maintenance workers pass behind the rack. The work was being performed under Maintenance Work Request (MWR) AP-3364.

SUPPLY SYSTEM POSITION

The Supply System concurs with each example violation.

ROOT CAUSE

In Examples 1 and 2 of the violation, the subject sign/barriers were apparently dislodged and knocked down during passage through barriers. The root cause is considered to be failure to follow procedures in that Health Physics was not notified of the down barriers. In Example 3 of the violation, the HRA was not adequately posted to preclude inadvertent entry. The root cause is inattention to detail when establishing the posting for the area.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

Based on the Supply System's investigation of these examples, there is no evidence of any unexpected and/or unnecessary personnel exposure. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

1. The immediate corrective actions were to establish or re-establish adequate sign/barrier postings at the locations.
2. A plant walkdown was performed to ensure all barriers were complete and the subject rope sign/barrier postings were subsequently replaced with swing gates.

3. PPM 11.2.7.1 was subsequently revised such that non-interactive barriers (such as swing gates) are used at ingress and egress points to radiation areas.
4. The Health Physics Tour Report Form was revised to give more specific direction on what to look for during daily posting inspections.
5. A Health Physics "Time Out" will be held by September 30, 1993, to discuss plant personnel responsibilities in maintaining sign/barrier postings.

E. SUMMARY: APPARENT VIOLATION 93-18-06

An NRC inspector reviewed overtime records for Operations personnel to determine compliance with the restrictions contained in Technical Specification (TS) 6.2.e.4. A guideline of TS 6.2.e.4 states that "[e]xcept during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on shift." On June 14, 1993, the Operations Division Manager generated an Interoffice Memorandum (IOM) which "extended a blanket overtime authorization up to the TS limit for the entire staff beyond the end date of the R8 outage." The inspector found 15 instances of overtime extension after R8 based on the blanket authorization (i.e., without individual consideration as specified in the TS).

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation, but does not believe that this concurrence constitutes an obligation for documented pre-approval of all overtime in excess of forty hours per week or eight hours per day.

ROOT CAUSE

On June 14, 1993, the Operations Division Manager issued the IOM to extend the waiver of administrative overtime limits to cover the time period between June 15 through July 15, 1993. As it turned out, the reactor was placed in Operating Mode 2 at 1635 hours on June 18, 1993. Control rod withdrawal began at 0230 hours on June 19, 1993, and the generator was synchronized to the grid at 1827 hours on June 21, 1993, ending the outage.

In hindsight, there were ways that the IOM likely could have been worded to preclude the blanket extension of overtime beyond R8. For example, the IOM could have indicated that its authorization would terminate when Mode 2 was entered or upon generator synchronization. In any case, the simplest method would have been to promptly cancel the extension. Therefore, the root cause of the violation is that the Operations Division Manager neglected to cancel the June 14, 1993 IOM following plant startup.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

All Operations overtime is either scheduled or requested by Operations management as outlined in both the Bargaining Unit Contract and Supply System General Information Handbook (GIH) 4.3.1, "Premium and Overtime Pay - Nonbargaining Employees." In addition, employee overtime is tracked and approved by the supervisor after it is worked. Based on an evaluation of Operations personnel timesheets, there is no evidence of anyone working in excess of seventy-two hours of overtime per week during the June 19 through July 15, 1993, period in question. The average number of overtime hours per Operations employee that worked overtime during this period was approximately eight hours per week. This indicates that the Operations staff was controlling and administering overtime in an appropriate manner despite blanket approval for up to seventy-two hours. In fact, "[t]he inspector noted that when personnel exceeded licensee administrative limits for overtime, procedures for obtaining authorization to exceed the limits were followed." Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

1. The immediate corrective action was for the Operations Division Manager to cancel the blanket overtime authorization. The Plant Manager verbally communicated to the NRC that blanket overtime authorization would not be allowed beyond the end of an outage in the future.
2. PPM 1.3.27, "Excessive Hours Worked Control," will be revised by October 1, 1993, to remove the administrative hours worked restrictions.

F. SUMMARY: APPARENT VIOLATION 93-18-07

An NRC inspector reviewed core alteration activities and noted several apparent errors. These errors include:

1. "On May 9, 1993, the operators skipped step 249 of the refueling sequence. Refueling bridge personnel discovered the error because, when performing step 250, personnel noted that the location in which the fuel bundle was to be placed was already full."
2. "On May 9, 1993, an error in the fuel handling procedure resulted in the skipping of a step that performed operations with the fuel prep machine."
3. "On May 12, 1993, a fuel bundle was misoriented 90 degrees. The refueling senior reactor operator on the next shift discovered the error by noting that the pattern of the fuel bundle bails was asymmetrical."

4. "On May 13, 1993, an error was discovered in the Nuclear Components Transfer Lists (NCTLs) in that the NCTL allowed any orientation for the dummy fuel cells... placed in the core; despite direction in PPM 6.3.2... to orient them in a prescribed manner. As a result, several cells... were improperly oriented. A Quality Assurance engineer discovered this error during a review of refueling procedures."
5. "On May 23, 1993, during the full core verification of fuel placement, a licensee engineer discovered that one fuel cell was misoriented by 180 degrees."

SUPPLY SYSTEM POSITION

The Supply System concurs with each example of the violation.

ROOT CAUSE

Example 1

During fuel movement, the Control Room phone talker inadvertently read Step 250 of the NCTL to the refueling RO before Step 249. The result was that a fuel bundle was removed from its core location to be placed in a spent fuel pool storage cell which was occupied. The purpose of Step 249 was to remove a fuel bundle so that the receiving cell was unoccupied.

During the fuel movement, it was quickly recognized by both the refueling Reactor Operator (RO) and Senior Reactor Operator (SRO) that the cell was occupied and, thus, refueling operations were immediately stopped. The bundle in transit was placed back into its original cell and Step 249 was successfully performed before reinitiating Step 250. A log entry of the incident was made in the Reactor Operator's Log and the Refuel Log.

The root causes were: (1) inattention to detail in the communication of critical information by the refueling phone talker and (2) a lack of independent verification by the refueling RO and SRO.

Example 2

While performing a channel swap pursuant to IOM SS2-PE-0066, the steps contained in the NCTL were not performed sequentially as required by PPM 6.3.28. Specifically, the NCTL for the fuel channel swap had a note on the cover sheet that stated "Do not leave fuel in Prep Machines at end of shift." However, the NCTL sequence did not provide any stop points where both Prep Machines were empty at the same time.



Due to a misunderstanding, the refueling SRO believed that only core alteration NCTLs had to be performed in strict order. Consequently, he skipped Step 38 to comply with the cover sheet note. Step 40, therefore, was the next step completed when work resumed. A formal deviation should have been initiated to correct the NCTL error in accordance with PPM 1.2.3, "Use of Controlled Plant Procedures," prior to continuation of the refueling activity.

The root causes were: (1) an inadequate NCTL and (2) failure by the refueling SRO to strictly follow refueling and administrative procedures.

Example 3

PPM 6.3.2 requires that during fuel shuffle the orientation of the bundle be properly established prior to insertion into the core. This is determined initially by the refueling RO and then verified by the refueling SRO. During the fuel shuffle for fuel bundle XN2091, the bundle was inserted into the core misoriented and was not recognized by the on-shift refueling RO or SRO. The misoriented fuel bundle was subsequently identified during the next shift by the on-shift refueling SRO. The root causes of this situation were: (1) a lack of attention to detail by the refueling RO and (2) a lack of independent verification by the refueling SRO.

Example 4

Although the proper blade guide orientation information was not included in the NCTLs, it was included in Attachment 8.4 of PPM 6.3.2, "Bridge Manipulations During Refueling." This attachment provides specific instructions required for refueling bridge operation and it is Operations management's expectation that the refueling ROs and SROs have a working knowledge of the contents of the attachment.

The root cause was a failure to meet expectations in that the refueling RO and SRO were unfamiliar with the blade guide orientation requirement specified in Attachment 8.4 of PPM 6.3.2. A contributing cause was the less than adequate NCTLs.

Example 5

This error is similar to the concern identified in Example 3 above. The root causes were: (1) a lack of attention to detail by the refueling RO and (2) a lack of independent verification by the refueling SRO.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

Example 1

The incident was promptly identified and corrected before any safety implications arose. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

Example 2

Fuel accountability was never lost and, thus, there was no impact on safety-related equipment or activities. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

Example 3

Immediately upon discovery of the misoriented fuel bundle, refueling activities were halted until approval could be obtained for an NCTL deviation to reorient the bundle. Because the misorientation was identified shortly after initial placement, the only impact on refueling activities was a short delay. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

Example 4

Fuel and blade guide accountability was never lost and the misoriented blade guides were correctly reoriented before any safety implications arose. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

Example 5

The misoriented fuel bundle was identified and correctly reoriented during the performance of PPM 6.3.5, "Full Core Verification." One of the procedural requirements is to specifically check the core for proper fuel bundle orientation so any misplaced bundle can be corrected before it causes power distribution problems. Thus, the fuel loading error was discovered during performance of the procedure that is designed to identify this type of error before any safety implications arise. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

The following corrective actions have been implemented for the incidents collectively:

1. The refueling phone talker was counseled on the importance of critical communications and attention to detail.
2. PPM 6.3.2 was revised to require the refueling SRO to directly supervise all movement of irradiated fuel.
3. The NCTL was revised to ensure that stop points resulted in empty Prep Machines.

4. The misoriented blade guides were reoriented to their proper orientation.
5. The Operations Manager held an "Operations Time Out" on May 10, 1993, to discuss refueling activity errors and other issues.
6. The refueling team was counseled on improving attention to detail and self checking.
7. The 90 degree misoriented fuel bundle was reoriented to its proper orientation.
8. The 90 degree misoriented fuel bundle error was discussed with refueling SROs. The necessity for independent verification of fuel movements was reemphasized.
9. The 180 degree misoriented fuel bundle was reoriented to its proper orientation.
10. Following discovery of the 180 degree misoriented fuel bundle, the core area was reverified for proper fuel loading.
11. Upon identification of these refueling errors, Supply System Quality Assurance commenced continuous monitoring of refueling operations on May 20, 1993.
12. At the completion of refueling operations, Supply System Quality Assurance performed an independent full core verification for proper fuel loading.
13. The refueling errors identified in Inspection Report 93-18 will be reviewed with refueling personnel prior to the movement of irradiated fuel during the 1994 refueling outage (R-9).

G. SUMMARY: APPARENT VIOLATION 93-18-09

Plant Modification Request (PMR) 92-0120 was initiated to downsize fuses on certain motor operated valves (MOV's). A project engineer scheduled the PMR work on the affected MOV's without first receiving Plant Operating Committee (POC) review and approval in accordance with the requirement in PPM 1.4.1, "Plant Modifications." Four fuse changes were implemented before receiving POC review and approval.

The inspector observed that the Supply System missed several opportunities to prevent the problem: (1) maintenance personnel did not recognize the implications of mixing corrective maintenance with design changes and (2) QC in their review of the PMR did not identify the lack of POC approval.

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation.

ROOT CAUSE

Because of the minor nature of the work contemplated by PMR 92-0120, the Project Engineer on this matter decided to add the BDC fuse replacement to existing MWRs. These MWRs had already been approved for diagnostic testing on twenty-three MOVs. The engineer believed that the addition of this simple design change to the existing MWRs was an efficient means of completing both tasks. However, it was not the normal method for implementing design changes.

The BDC package was provided to the MWR preparer by the project engineer. The preparer revised the MWR to add a step to the instructions that stated: "Replace the MCC 480 volt power fuses . . . per BDC 92-0120-1B." This change in the instruction was signed off by a Maintenance Engineer, a Maintenance Supervisor, and a Quality Control Engineer.

Only four MWR packages with the fuse replacements were actually completed before the Project Engineer became aware that the PMR had to be approved by POC before implementation of the BDC. The engineer promptly initiated PER 293-321 on March 24, 1993.

The Project Engineer in this event was new to his position and had never been involved with processing a design change at the Supply System. His prior experience in Engineering did not include preparation of MWR work instructions for implementation of BDCs. Since he was new and did not have a working knowledge of PPM 1.4.1, it is management's expectation that he should have had the procedure available and consulted it for direction.

Concerning the Maintenance Engineer and the Maintenance Supervisor, they were both long-time employees of the Supply System and had received training in such matters. However, when the Project Engineer presented an apparent change to the normal process, they did not consult the governing procedures.

The QC requirement to verify the PMR cover sheet on MWRs implementing design changes is specified for the initial review of the package. When reviewing changes to the package, the QC Engineer only reviews the change, as there is no requirement to review documentation specified for the initial review. Based on these defined responsibilities, the fuse replacement revisions (changes) to the MWRs were reviewed and assigned hold points without verifying POC approval on the PMR cover sheet. The QC Engineer reviewing the MWR changes failed to recognize that the original MWRs had not been initially reviewed as design changes. Thus, the initial QC reviews did not include the PMR and the subsequent change reviews did not verify the PMR. Consequently, POC approval on the PMR cover sheet was never verified during the QC review process.

The root cause was inattention to detail in implementing the relevant procedures.



SIGNIFICANCE & MITIGATING CIRCUMSTANCES

This violation was discovered by the responsible Project Engineer and promptly reported in PER 293-321. Replacement of the four MOV fuses prior to approval of the PMR did not impact the safety or operation of any plant equipment. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

The following corrective actions were implemented:

1. PMR 92-0120-01 was approved by POC on April 7, 1993.
2. The four MWRs that were previously worked were revised to add the requirements of PPM 1.4.1 and PPM 1.3.7, and the Control Room top-tier drawings were red-lined. The remaining nineteen MWRs were revised accordingly.
3. Project Engineers were reminded of their responsibility for assuring that the requirements of PPM 1.4.1 are satisfied when processing design changes.
4. Maintenance Engineers were reminded of their responsibility for assuring that the requirements of PPM 1.3.7 are satisfied when processing design changes.
5. A memorandum was issued to all QC inspectors describing the event and defining their responsibilities during initial review and review of changes to MWRs implementing design changes.

H. SUMMARY: APPARENT VIOLATION 93-18-10

During the R8 outage, "Type C" "as-found" local leak rate testing (LLRT) was not performed on valves RHR-V-16B and RHR-V-17B. 10 C.F.R. Part 50, Appendix J, requires "Type C" leakage testing. In a letter dated April 29, 1987, the NRC granted certain exemptions from Appendix J to the Supply System with "several stipulations," including that "Type C testing be performed prior to any maintenance or repair of the penetration barrier." This requirement is reflected in PPM 7.4.6.1.2.4.

When an NRC inspector questioned the testing coordinator (a Shift Technical Advisor [STA]) about the LLRT, the STA indicated that the testing was not required. The Plant Technical Supervisor subsequently informed the inspector that as-found LLRT was required and that, independent of the inspector's questioning, the Supply System had determined that such testing had not been performed.

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation.

ROOT CAUSE

An outage integrated valve schedule was issued on April 4, 1993, for the 1993 refueling outage (R-8). This schedule included only "as-left" Local Leak Rate Testing (LLRT) for motor operated Residual Heat Removal (RHR) boundary valves RHR-V-16B and RHR-V-17B since no maintenance or testing activities were scheduled for the valves. However, on May 3, 1993, refurbishment of these valves was added to the schedule without adding an "as-found" LLRT. The LLRT Coordinator was not made aware of the schedule change. Thus, when he was contacted by the Motor Operated Valve (MOV) Refurbishment Coordinator and asked "if the RHR System B 'as-found' LLRTs had been completed," he responded affirmatively, not realizing that this testing was to have included RHR-V-16B and RHR-V-17B. The MOV Refurbishment Coordinator then used this nonspecific response to sign off the notification prerequisites for the RHR-V-16B and RHR-V-17B refurbishment Maintenance Work Requests (MWRs). This action authorized work to commence on the valves without the "as-found" LLRTs having been performed.

The root cause was less than adequate verbal communication, in that the MOV Refurbishment Coordinator failed to specify "RHR-V-16B" and "RHR-V-17B" when confirming completion of the associated MWR prerequisites.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

The failure to perform "as-found" LLRTs on RHR-V-16B and RHR-V-17B temporarily eliminated them from the 10CFR50 Appendix J Exemption provisions. The only impact on the plant is that "Type C" LLRTs will have to be performed on the valves during the 1994 refueling outage (R-9). Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

1. Discussions were held with the LLRT Coordinators (including the STA who provided the incorrect valve testing requirement information to the NRC inspector), the Outage Coordinators, and the MOV Refurbishment Coordinator stressing the importance of using the Equipment Piece Numbers (EPNs) when communicating with the LLRT Coordinator. In addition, the discussions clarified the differences between LLRT hydraulic tests and 10CFR50 Appendix J tests with respect to "as-found" testing.
2. "Type C" LLRTs will be performed on RHR-V-16B and RHR-V-17B during the 1994 refueling outage (R-9).

3. PPM 1.3.60, "Verbal Communications Policy," will be issued by October 15, 1993, to establish appropriate methods for plant communications.

I. SUMMARY: APPARENT VIOLATION 93-18-12

During performance of the vendor recommended, six-year preventive maintenance on the Division 2 Diesel Generator, an NRC Inspector questioned whether an air pressure test was performed using a calibrated gage. The work involved replacement of seals (i.e., gaskets) used on the engine cylinders, in accordance with MWRs AP1184 and AP1186. Step 2.40 of each MWR called for a pressure test which necessitated a special test rig. However, the gage used with the test rig had not been calibrated. Operational Quality Assurance Program Description (OQAPD), Section 12, "Control of Measuring and Test Equipment," and PPM 1.5.4, "Control of Measuring and Test Equipment - Transfer Standards," require that a calibrated gage be used in activities affecting quality.

When questioned by the NRC Inspector about the matter, the QC inspector indicated that a calibrated gage was not required because the testing was for "information only" (i.e., qualitative rather than quantitative).

SUPPLY SYSTEM POSITION

The Supply System does not agree with the violation as stated in the Inspection Report. However, the Supply System admits a violation of PPM 1.5.4, Step 6.3.3.

ROOT CAUSE

During the preparation for testing, and during the actual testing, the involved Maintenance Engineer, the Diesel Generator System Engineer, and two different QC Engineers considered the test an "information only" test. Their reasoning was based on the fact that PPM 1.5.4, Section 4.4, "Non Calibrated M&TE [Meter and Test Equipment], Non Data," allows the use of noncalibrated instrumentation when the data to be gathered is not for quantitative purposes. The pneumatic test of the diesel engine cylinder head and liner seals is considered "good shop practice" at all General Motors Electromotive Division (EMD) engine shops for qualitative purposes (to identify a damaged or defective seal). This test is not required by any regulatory requirements or by the engine manufacturer. Safety-related acceptance testing for the diesel generators is specified in Regulatory Guide 1.108. Therefore, a violation of OQAPD, Section 12, did not occur and the violation of PPM 1.5.4 was not as specified in the Inspection Report.

The Supply System's review of this matter indicates that a different noncompliance of PPM 1.5.4 occurred. Plant personnel involved failed to fully comply with Section 6.3.3 of the procedure by not labeling the noncalibrated test gage "Non Data." This procedural requirement was found to be a little known and unenforced provision. The majority of plant personnel are unaware of the provision and it is not acknowledged in the work

process (e.g., PPM 1.3.7, "Maintenance Work Request"). Plant maintenance personnel have routinely used noncalibrated test equipment (e.g., tool bag volt-ohmmeters) for "information only" testing without "Non Data" labels. The root cause was less than adequate management methods. The M&TE Program failed to adequately implement the "Non Data" labeling requirement of PPM 1.5.4, or recognize it as unnecessary and remove it. A contributing cause was a personnel work practices error in that plant personnel technically failed to comply with the procedure.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

The failure to label the noncalibrated test gage "Non Data" had no impact on diesel generator acceptance testing as specified in Regulatory Guide 1.108. Therefore, the Supply System believes the incident is of minimal safety significance, which is consistent with the findings of Inspection Report 93-18.

CORRECTIVE ACTIONS

1. The immediate action was to obtain a calibrated gage and perform the air testing again.
2. As part of the disposition of PER 293-0637, a meeting was held on June 3, 1993, with the appropriate Supply System personnel to discuss "information only" testing and establish a plant policy. The findings and recommendations of the meeting were communicated to affected managers and supervisors.
3. Appropriate M&TE personnel will be trained on the procedural requirements and provisions of PPM 1.5.4 by October 15, 1993.
4. PPM 1.5.4 will be revised by October 15, 1993, to remove the requirement to label noncalibrated test equipment as "Non Data" since calibrated test equipment must be labeled as such.

ATTACHMENT 2

RESPONSE TO APPARENT VIOLATIONS

CITED IN NRC INSPECTION REPORT 93-24

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NRC INSPECTION REPORT 93-24: RESPONSE TO APPARENT VIOLATIONS

SUMMARY: APPARENT VIOLATION 93-24-02

Technical Specification paragraph 6.8.1.g states that written procedures are to be implemented for the Fire Protection Program. Plant Procedures Manual (PPM) 1.3.10, "Fire Protection Program," states in Paragraph 6.3.5.a, "...combustible liquids must be removed and put into storage at the end of the job or at the end of the shift if the job is not continuous between consecutive shifts." Paragraph 6.3.8.a of PPM 1.3.10 states, in part, "...when removal is not possible, a Transient Combustible Permit is required if the combustibles are to be left unattended for any length of time (i.e., breaks, lunch)."

During a tour of the RHR A pump room on July 17, 1993, an NRC inspector found a five-gallon bucket of lubricating oil on the floor. No personnel were in the vicinity and a "Transient Combustible Permit" was not present. Further investigation revealed added significance for this problem. The RHR A pump room was on the hourly fire tour. Approximately 48 fire tours by three different individuals were conducted in the RHR A pump room while the combustible material was present prior to the inspector entering the area. This failure is of concern because Supply System management had previously shared their expectation during an enforcement conference conducted on June 29, 1993, that fire tours would not be cursory and indicated that this view had been clearly communicated to the plant staff.

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation. The maintenance personnel that drained the oil should have removed it when the job was completed. The implication that the fire tours should have identified this transient combustible is not as clear cut. The fire impairment which necessitated the fire tour in this area is within twenty feet of the door used for normal access. The area just inside this door gives the personnel performing the fire tour a clear view of the impairment. Thus, the Supply System believes the fire tour personnel performed their inspection of this area as expected and within requirements.

At the June 29, 1993, enforcement conference the focus was on the conduct of tours. Specifically, that the operator's initials on the log indicated he personally had entered the space when a fire tour was required and evaluated the space to ensure signs of smoke and fire were not present. The issue of what items were to be observed when conducting fire tours was not discussed during the enforcement conference.

ROOT CAUSE

On July 17, 1993, nonlicensed electrical maintenance personnel repaired a leak in the RHR Pump A upper bearing oil reservoir sight glass. Approximately three gallons of oil were drained into a plastic bucket to make the repair. New oil was used to replace the amount in the upper oil reservoir. The old oil was left behind in the vicinity of the step off pad when the job was completed.



The root cause of this event was improper work practices in that self checking was not applied to make sure the oil was removed prior to leaving the area.

SIGNIFICANCE & MITIGATING CIRCUMSTANCES

The event by itself is not significant as the amount of oil was small and it is believed this was an isolated occurrence. This is consistent with the findings of Inspection Report 93-24.

CORRECTIVE ACTIONS

1. The electrical maintenance crew and their supervisor have been counseled.
2. A "Lessons Learned" session was held with Electrical Shop Supervision and the Maintenance Production Manager. This stressed the requirements of PPM 1.3.10, the need to self check and maintain attention to detail.
3. Management expectations for fire tour personnel have been clarified. This includes expectations with respect to combustible loading, hot work, and impairments.

SUMMARY: APPARENT VIOLATION 93-24-05

On June 30, 1993, after the reactor was in Mode 1 and at 100% power, an NRC inspector reviewed the Temporary Modification Request (TMR) log in the control room to determine if it had been cleared of outage-related temporary modifications. The inspector found TMR 93-017 in effect when it should have been removed prior to plant startup. TMR 93-017 installed temporary leads on the Scram Discharge Volume (SDV) Vent and Drain Valves to assist with stroke time testing of the valves.

The temporary leads were removed from the plant on June 18, 1993, prior to plant startup as required by the TMR. However, several administrative procedures were not followed as part of this removal process:

1. The "Work Supervisor" did not comply with PPM 1.3.9, Temporary Modification Requests, Sections 6.3.6, that requires removal of TMR tags, signing of the TMR form, and notification to the Shift Manager that the temporary modification has been removed.
2. The Shift Manager did not follow PPM 1.3.9, Section 6.3.7, that requires the completion of six steps when a temporary modification is restored. In addition, Section 7.4.1 requires the Shift Manager and the Control Room Supervisor to review the temporary modification log every shift as part of their shift turnover process.
3. The Operations Manager did not implement the PPM 1.3.9 Section 7.4.3 requirement to review the TMR log prior to plant restart. In addition, Section 7.4.4, requires a weekly review to remove completed TMRs.



4. The Technical Services Manager did not implement PPM 1.1.7, Restart Evaluation Process, that requires a review of the TMR log prior to plant restart.

These failures could lead to less than adequate knowledge of plant conditions by licensed operators and plant management.

SUPPLY SYSTEM POSITION

The Supply System concurs with the violation. TMR 93-017 was written by the System Engineer to support Instrumentation and Control work on SDV Vent and Drain Valves CRD-V-10, 11, 180, and 181. The TMR was initiated during plant shutdown on May 1, 1993, to allow previously installed test leads to the limit switches on these valves to remain in place until the end of the outage. Leaving the test leads in place avoided a significant amount of radiation exposure (200-300 mrem) since they were known to be required to support valve testing at the end of the outage. The TMR utilized one-tag that was hung on the bundle of eight test leads at the back of the chart recorder used to measure valve position.

On June 18, 1993, valve testing was successfully performed per Plant Procedure, PPM 7.4.1.3.1.1B, Scram Discharge Volume Vent and Drain Valves Quarterly Operability. Further investigation showed that permanent plant I & C personnel were used to operate the chart recorder and collect the stroke time data following the outage. However, because of the significant dose exposure required, two temporary I & C technicians were utilized to remove the leads. The use of multiple crews may have impacted the closeout of TMR 93-017. Further, PPM 7.4.1.3.1.1B, step 11 requires signoff and verification that the chart recorder and test leads were removed. This may have caused personnel performing the test to overlook the administrative requirements of the temporary modification.

The Supply System believes some of the statements in the apparent violation need clarification. The Shift Manager was not able to perform PPM 1.3.9, Section 6.3.7, since he was not notified by the Work Supervisor of the removal of the TMR. The PPM 1.1.7 Restart Evaluation Process is performed before actual plant startup and involves a functional review of the status of the TMRs. In this case, the restart review of outstanding TMRs was performed on June 14, 1993. TMR 93-017 was listed as part of the review and it was noted it would be "...cleared after testing prior to startup."

ROOT CAUSE

The root cause of this event was less than adequate work practices. The workers and supervision should have recognized the significance of the TMR tag and taken appropriate action to close out the TMR. Likewise, the Operations Department reviews of the TMR log in the Control Room should have identified this discrepancy.



SIGNIFICANCE & MITIGATING CIRCUMSTANCES

The event by itself is not significant as the temporary modification was physically removed prior to plant startup. This is consistent with the findings of Inspection Report 93-24. However, the Supply System is disturbed by the fact that multiple barriers placed in the procedures failed to detect this condition.

CORRECTIVE ACTIONS

1. Maintenance and Operations personnel having responsibility to implement procedures associated with TMR 93-017 will be counseled on lessons learned from this event. This will be completed by September 15, 1993.
2. An evaluation will be performed to determine how the ownership of a TMR can be improved. This will be completed by December 1, 1993.



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