ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-267/95-01

License: DPR-34

Licensee: Public Service Company of Colorado
P.O. Box 840
Denver, Colorado 80201-0840

Facility Name: Fort St. Vrain Nuclear Generating Station

Inspection At: Fort St. Vrain, Platteville, Colorado

Inspection Conducted: February 6-9, 1995

Inspectors: R. J. Evans, Health Physicist
Fuel Cycle and Decommissioning Branch

Charles L. Cain, Chief
Fuel Cycle and Decommissioning Branch

Approved: Charles L. Cain — 3/6/95
Charles L. Cain, Chief — Date
Fuel Cycle and Decommissioning Branch

Inspection Summary

Areas Inspected: Routine, announced inspection of facility status, operational safety verification, occupational exposure during decommissioning, and followup of previously identified inspection findings.

Results:

- The core support floor was lifted during the inspection period. The lift appeared to be a well planned, controlled, and executed activity (Section 1).

- NRC permission to discontinue weekly notifications to report waste water permit violations was administratively cancelled when the State of Colorado amended the associated waste water discharge permit (Section 2.1).

- The radiation monitoring equipment was inspected and was found to be in operation, indicating licensee compliance with Decommissioning Technical Specification 3.3 requirements (Section 2.2).

- During routine tours of the facility, housekeeping in the turbine building and independent spent fuel storage installation were found to be acceptable. Housekeeping in the reactor building needed improvement (Section 2.3).

- The control of radiological postings and boundaries was determined to be acceptable although several minor implementation problems were identified (Section 2.3).

- A repeat violation was identified involving the failure to adhere to radiation protection procedure requirements for radioactive material labelling (Section 2.3).

- Increased levels of radioactive materials were identified in the plant's effluent pathway. The material apparently was introduced to the environment during recent decommissioning activities (Section 2.4).

- The licensee experienced its first positive bioassay incident during decommissioning in December 1994. The amount of radioactive material that was ingested by the individual was determined to be negligible (Section 3.1).

- The licensee's activities for final surveying of the repower area were reviewed. An NRC confirmatory survey is planned (Section 3.2).

- The licensee had established an acceptable Radiation Work Permit (RWP) Program. Several minor procedure discrepancies and implementation concerns were identified and were reported to the licensee (Section 3.3).

<u>Summary of Inspection Findings</u>:

- Violation 50-267/9501-01 was opened (Section 2.3).

- Deviation 50-267/9302-01 was left open (Section 4.1).

- Violation 50-267/9402-01 was closed (Section 4.2).

- Unresolved Item 50-267/9403-02 was closed (Section 4.3).

- Violation 50-267/9406-01 was closed (Section 4.4).

<u>Attachment</u>:

- Persons Contacted and Exit Meeting

DETAILS

1 FACILITY STATUS

Decommissioning of the Fort St. Vrain facility was being performed by a Westinghouse Team that consisted of personnel from Westinghouse Electric, Scientific Ecology Group (SEG), and MK-Ferguson. The major decommissioning task in progress at the facility was the dismantlement and decontamination of the radioactive portions of the prestressed concrete reactor vessel.

The removal of the core support floor was the next major evolution in the decommissioning process. The core support floor was a 5-foot thick disk that was 29 feet in diameter and weighted roughly 270 tons. The floor was constructed of reinforced concrete within a steel enclosure. The floor was supported by and welded to 12 steel columns. Also, 12 vertical penetrations were located between the columns and were lined with steel ducts leading to the inlet of the steam generators. The floor removal work consisted of: (1) installing multiple lifting jacks and support equipment, (2) detaching the core support floor from the steam generator inlet ducts and core support floor columns, (3) raising the core support floor about 45 feet, (4) installing support beams below the core support floor on the lower hexagonal ledge, (5) lowering the floor a few feet to allow the floor to sit on top of the support beams, (6) cutting the floor into two major pieces, (7) transferring the pieces to a containment building with the overhead crane for sectioning, (8) cutting the floor with a diamond wire saw into transferable sections, and (9) shipping the floor offsite for disposal.

Support steel and jacking mechanisms were installed for the 343 ton lift. The weight was a conservative estimate that included attached components such as a monorail system, water soaked insulation, personnel shielding, and an extra 10 percent margin of safety. The jacking mechanisms consisted of four banks of three jacks each. Each of the four jacking stations were rated at 105 tons.

The decommissioning safety review committee reviewed and subsequently approved the radiation work permit on February 7, 1995, for the core support floor lift, floor disassembly and shipment, and associated support work. The exposure estimate for the work was broken down into sub-tasks with a total estimate of 122 person-rems for the job, although the actual exposure may be less. The project total was originally estimated to be 433 person-rems with the core support floor removal being the second highest exposure job at 105 person-rems (the highest was the dismantlement of the reactor core and core barrel).

The core support floor was lifted on February 8, 1995. Overall, the core support lift appeared to be a well controlled and planned evolution, as evidenced by the completion of the task with a minimal number of impedances. RWP restrictions, such as continuous air sampling for radioactive contaminants, were adhered to by the support personnel. The full weight of the core support floor was placed on the six support beams on February 13, 1995.

A large containment building was being installed on the south end of the refueling floor for the core support floor sectioning work. The containment will be used to contain radioactive contaminants during the cutting of the floor for offsite shipment. Other support equipment being installed included a ventilation system, the diamond wire cutting system, and a collection system for the slurry that will be generated during the cutting evolution. The core support floor lift, cut, and support work were expected to continue until May or June 1995.

Other work in progress during the inspection included the installation of thermoluminescent dosimeter (TLD) strings and pulleys. The licensee plans to install strings of TLDs 20 inches apart in balance of plant system piping for at least 10 weeks to measure the exposure rates in the pipes. The licensee plans to use the TLD strings to obtain exposure data at about 5000 points.

Future activities planned included removal of the steam generators and helium circulators, performing additional beltline concrete cuts in the prestressed concrete reactor vessel cavity, and removal of the core support columns and lower plenum floor.

The completion of the decommissioning project is currently scheduled for June 1, 1996.

2 OPERATIONAL SAFETY VERIFICATION (NRC Inspection Procedure 71707)

The purpose of this inspection was to ensure that decommissioning activities were being conducted safely and in conformance with license and regulatory requirements. The following paragraphs provide details of specific inspector observations during this inspection period.

2.1 Deletion of pH Limitations and Monitoring Requirements For Farm Pond Effluent

In a letter dated October 25, 1994, to the Region IV Regional Administrator, the licensee requested permission to discontinue the weekly notifications required by 10 CFR Part 50.72 for high pH in the farm pond effluent (part of the effluent pathway from the plant's restricted area to the environment). The weekly call to the NRC Operations Center was required each time the licensee notified the State of Colorado's Department of Health and Environment of a violation of their waste water discharge permit. The NRC granted the licensee permission by letter dated November 16, 1994, to discontinue the weekly notification to the NRC with the understanding that the State planned to modify the waste water discharge permit in the near future to delete the limits for farm pond pH. On February 1, 1995, the licensee's waste water permit, as amended by the State to delete the limitations and monitoring requirements for pH in the farm pond effluent, became effective.

2.2 Radiation Monitoring Instrumentation System Walkdown

In accordance with Decommissioning Technical Specification 3.3, the radiation monitoring instrumentation channels are required to be operable at all times. The instrumentation consists of two monitors, one located on the refueling floor and the second in the truck bay of the reactor building. These monitors serve as accident monitors to detect unplanned radiation levels in the reactor building. During routine tours of the reactor building, the two channels were inspected to ensure that the monitors were in service and capable of performing their intended functions. The monitors were found to be operable.

The refueling floor monitor was located on the south wall of the reactor building. Between the monitor and the reactor vessel cavity, the licensee's contractors were installing a containment building for the core support floor cut. The building consisted primarily of reinforced sheet metal and plastic sheets. The contractor plans to install some shielding to protect the workers (an ALARA concept) when the core support floor is transferred to the containment building. The shielding installation will have to be controlled to ensure that the effectiveness of the radiation monitor is not reduced.

2.3 Facility Tours

Routine tours of the reactor building and other areas of the plant were performed to determine if the facility was being decommissioned in accordance with the license, Decommissioning Plan, and regulatory requirements. Specific attributes that were inspected included the maintenance of the radiologically controlled areas, housekeeping, and material control. Also inspected was the control of radiological postings, labelling, and boundaries.

Housekeeping was previously rated as good in the reactor building and poor in the turbine building (refer to NRC Inspection Report 50-267/94-06). The reverse appeared to be true during this inspection. General housekeeping had improved significantly in the turbine building while housekeeping in the reactor building needed improvement. Housekeeping in the independent spent fuel storage installation (ISFSI) was also considered to be adequate. Some improvement was noted in the reactor building during the course of the inspection. A contribution to the problem was the high number of bagged items in the reactor building that had been set aside for future decontamination or reuse. Heavy dust accumulation was present in many areas of the reactor building. The prestressed concrete reactor vessel pipe cavity air handling unit Fan 1B was in service during the inspection. The air handling unit's inlet filters were noted to be clogged with dust and other fine particles and appeared to need cleaning. Although this air handling unit is no longer required to be operable, the failure to maintain clean, unobstructed filters could have a negative effect on the ability of this component to perform its intended function.

Radiological boundaries were well defined and marked off. Two radiological postings were identified that did not strictly adhere to the requirements of the access control program procedure. The postings failed to identify the specific type of radiologically controlled area, such as "contaminated area," that required a radiological boundary. The two postings were a small percentage of the total postings in the reactor building and appeared to be isolated cases. A third posting was found lying on the floor. The procedure for operation and testing of portable ventilation equipment stated that equipment should not be operated unless the system contained filters and the filters are sealed. One operating prefilter container in the reactor building was latched but was not sealed. These findings were indicative of a lack of attention to detail on the part of the radiation protection personnel.

Radioactive Material Control Program procedure FSV-RP-RAM-A-100, Revision 3, Step 5.3.2, states that radioactive material shall be labelled "Caution (or Danger) - Radioactive Material" and provide sufficient information to alert personnel to the potential hazards of the material, such as contamination levels, contact radiation levels, and radionuclides present. Similar statements are provided in the Decommissioning Plan Section 3.2.6.2, Identification of Radioactive Material, and 10 CFR Part 20.203 for containers of radioactive materials. During routine tours of the reactor building, several components were found that were simply labelled "Caution-Radioactive Material" without providing sufficient information to alert personnel to the potential hazards of the material. Examples that were identified (and pointed out to licensee representatives) included tool boxes, protective clothing containers, and in-service and stored equipment.

During an inspection conducted in March 1994, several examples of a failure to adhere to radiation protection program procedures were identified. Violation 50-267/9402-01 was subsequently issued for the procedure adherence violations. One example cited involved the failure to properly label a contaminated sling in accordance with the requirements established by the radioactive material control program procedure. Corrective actions were described in the licensee's letter dated June 8, 1994, and consisted of revision of the labelling procedure, development of a training module, technician training, and building walkdown to identify additional examples of improper labelling. The corrective actions taken in response to the violation were apparently ineffective because deficient radioactive material labelling was again identified during this inspection. Contributors to the current problem include the high number of items labelled as radioactive material in the reactor building and the fact that more material is created and labelled on a daily basis.

Since the components were located in a radiologically restricted area and since all personnel and components have to be scanned for potential contamination prior to exit or release from the reactor building, a potential health and safety hazard did not exist because of the inadequate labelling.

License Condition 2.D(2) for License DPR-34, states that the licensee shall maintain the facility in accordance with the Technical Specifications. Decommissioning Technical Specifications Section 5.4.1 states, in part, that "written administrative procedures, plans, manuals, and/or programs shall be established, implemented, and maintained covering the activities referenced below: a. radiation protection program...." Section 5.7 states "procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all activities involving personnel radiation exposure." The failure to comply with the requirements of the Radioactive Material Control Program procedure FSV-RP-RAM-A-100, Revision 3, was identified as a repeat violation of the Decommissioning Technical Specifications referenced by License DPR-34 Condition 2.D(2) (50-267/9501-01).

Immediately after the conclusion of the inspection, the licensee provided the NRC with an update of their short-term corrective actions. A thorough walkdown of the reactor building was performed by SEG personnel. Of the roughly 1250 items labelled as radioactive materials, about 55 items had questionable labels, suggesting that less that 5 percent of the components were improperly labelled. Short-term corrective actions planned included retraining, implementation of weekly walkdowns and spot checks by radiation protection supervisors, and cleanup of selected areas of the reactor building.

2.4 Discovery of an Increase in Radioactive Material Levels in Plant Effluent Pathway

The licensee informed the NRC inspectors about their discovery of an increase in the amount of radioactive materials in the plant effluent pathway. During normal effluent releases from the plant, diluted liquid effluent is discharged from the restricted area to the Goosequill Ditch. From the Goosequill Ditch the liquid flows to the Jay Thomas Ditch, then discharges into a 25 acre farm pond. The pond discharge is then routed to the South Platte River.

Radioactive liquid effluents, from either the radioactive liquid waste system or the reactor building sump, are routinely diluted by the cooling tower blowdown flow prior to release to the surrounding surface waters. The rate of liquid release is normally controlled to assure that the radioactive material concentrations do not exceed 10 CFR Part 20 limits and that the tritium concentrations do not exceed EPA safe drinking water standards. The use of surface water downstream of the site is limited almost entirely to irrigation.

As part of the decommissioning process, roughly 100,000 gallons of shield water system volume were recently released to the environment via the normal plant effluent pathway. Routine samples were taken in the plant effluent pathway by SEG personnel following the release. Preliminary sample results indicate that elevated levels of cesium-137 and cobalt-60 were identified in the sediment of the two ditches. The sample results indicated that about 4 to 5 picocuries/gram of radioactive material were present in the sediment, with a background level of 1 picocurie/gram or less. Cattle ranching is the primary farming activity in the immediate vicinity of the plant. Environmental samples of milk and beef were taken in December 1994 and no elevated radioactive materials were identified in the samples.

Guidelines for acceptable levels of contamination in soil to be released for unrestricted use have been established by the NRC for both cesium-137

(15 picocuries/gram) and cobalt-60 (8 picocuries/gram). Since the radionuclides identified in the ditches do not exceed the release guidelines, an environmental health and safety concern did not exist. The licensee is considering generating a radiological occurrence report for the findings because one of the recommended criteria for a report was "radioactive materials found in uncontrolled areas." The licensee is required to account for the increased concentrations in the final survey report and the annual environmental monitoring submittal to the NRC.

2.5 Conclusions

NRC permission to discontinue weekly notifications to report waste water permit violations was administratively cancelled when the State of Colorado amended the permit.

The radiation monitoring equipment was inspected and was found to be in operation, indicating licensee compliance with Decommissioning Technical Specification 3.3 requirements.

During routine tours of the facility, housekeeping in the turbine building and ISFSI were found to be acceptable. Housekeeping in the reactor building needed improvement.

Overall, the control of radiological postings and boundaries was determined to be acceptable although several minor lapses were identified. A repeat violation was identified involving the failure to adhere to radiation protection procedure requirements for radioactive material labelling.

Increased levels of radioactive materials were identified in the plant's effluent pathway. The material apparently was introduced to the environment during recent decommissioning activities.

3 OCCUPATIONAL EXPOSURE DURING SAFSTOR AND DECON (83100)

Selected aspects of the radiation protection program were inspected to independently determine the adequacy of the program during decommissioning.

3.1 Internal and External Exposure Control

During the inspection, Radiological Occurrence Report 94-72 was reviewed and discussed with the licensee. The Report describes an event that occurred on December 7, 1994, where a site worker experienced facial contamination followed by a positive whole body count. While attempting to remove some bolts from underneath a work platform, a small amount of contaminated dust settled on the worker's face. The dust apparently was generated by workers on top of the platform. Contamination was identified on the worker during a routine body frisk. The worker washed his face and subsequently passed the body frisk check. The worker was directed to get a whole body count as a precaution.

The whole-body count was positive, indicating that the worker had ingested some radioactive material. The whole-body count showed a 2 percent body burden of cobalt-60. A second count was taken the next day and the results were still positive. A third whole-body count was taken on December 12, 1994, and the results were negative. The licensee conservatively concluded that the worker ingested 23.4 nanocuries of radioactive material. An evaluation was performed that determined that the worker experienced a committed effective dose equivalent of 0.24 millirems.

A radiological occurrence report was issued to investigate the event. Preliminary investigations reveal that no procedure or RWP restrictions were violated. Proposed corrective actions included improved communications between workers and radiation protection personnel. Although the amount of radioactive material ingested and the exposure received from the material were negligible, this incident was the project's first event that resulted in a positive bioassay finding.

3.2 Status Update of the Final Survey of the Repower Area

As part of the licensee's plan to convert the Fort St. Vrain facility to a natural gas powered power plant, the licensee recently cleared out about 5 acres of land in the southeastern corner of the restricted area. The licensee plans to start installing repower equipment in this area once the area is released by the NRC from the license. As part of the release process, the area was surveyed for radioactive contamination in January 1995. Also, background sampling was performed offsite for use in the final survey documentation for comparison to onsite sample results. The licensee's contractor, SEG, recently completed the sampling. The following samples were taken in the repower area:

- 62 surface and subsurface soil samples
- 185 gamma measurements at a one meter height
- 204 fixed activity measurements
- 12 well water samples from 3 wells
- 204 beta-gamma loose activity smear surveys
- 18 septic sludge-water samples
- 442 beta-gamma scan surveys

The following samples were taken at background locations:

- 104 surface and subsurface soil samples
- 191 gamma measurements at a one meter height

- 150 fixed activity measurements
- 12 well water samples from 4 wells
- 150 beta-gamma loose activity smear surveys
- No septic sludge-water samples

The licensee plans to submit the results of the repower area surveys to the NRC in the near future. The NRC plans to perform a confirmatory survey in late-March 1995 with the assistance of representatives from the Oak Ridge Institute for Science and Education.

3.3 RWP Program Review

The Decommissioning Plan, Section 3.2.5.4, described the requirements for RWPs. RWPs are used for the administrative control of personnel entering or working in areas that have, or potentially have, radiological hazards present. Administrative Procedure FSV-RP-OPS-A-300, Revision 2, "Radiation Work Permit Program," provided the requirements and guidance to implement the RWP program. Implementing Procedure FSV-RP-OPS-I-301, Revision 5, "Radiation Work Permit Preparation, Review and Approval," described the process for writing, maintaining, and terminating RWPs. During the inspection, the administrative and implementing procedures, 12 RWPS, and the associated surveys were thoroughly reviewed.

During the review of the administrative and implementing procedures, several conflicting statements were identified between the procedures. The implementing procedure provided instructions to post a copy of an RWP and the most recent survey at the main access control point into the reactor building and at the control point established for the job location. The administrative procedure provided similar instructions but the posting was required at the main access control point or at the control point established for the job location (this procedure did not require postings at both locations). The radiochemistry laboratory survey was found at the job location but not at the main access control point. That is, the survey posting was in compliance with the administrative procedure but not in compliance with the implementing procedure. Other procedure discrepancies included the level of approvals required for RWP revisions (significant versus non-significant changes) and whether the RWPs should be stamped as "copy" or "working copy."

The administrative procedure stated that the expiration date of an RWP could be extended but failed to indicate who had authority to grant extensions. This authority should have been clearly stated because the procedure implies that non-radiation protection personnel can authorize the extension.

Attachment 6.6 of the implementing procedure, Radiation Protection Support Guidelines, stated that continuous health physics coverage is required for any activity that included areas with contamination above 100,000 disintegrations per minute for any activity. This requirement was not incorporated into several RWPs that were reviewed which authorized access to high contamination areas. No example of an individual actually entering such an area without health physics coverage was identified.

An emergency RWP was developed for use in emergency conditions, such as a fire or medical emergency. The administrative procedure stated that all qualified radiation protection technicians should review and sign the RWP signature sheet acknowledging that they understand the use of the RWP. The procedure also stated that if radiation protection personnel were not on site (a condition that occasionally occurs during the graveyard shift or major holidays), then control room operations personnel shall perform radiation protection actions. The emergency response RWP No. 9999 was reviewed. The inspector noted that only 36 signatures were on the signature sheet (not all personnel were radiation protection personnel) although there were 42 technicians assigned to the project. Also, the procedure required that if the operations personnel are required to perform radiation protection personnel functions during emergencies, then the operations staff should review and sign the RWP also (none of the operations staff was identified on the signature page). Finally, the administrative procedure listed several restrictions that are necessary during an emergency; however, these restrictions were not carried over to the instructions section of the emergency response RWP.

Each RWP is required to have a current copy of the most recent survey with or near the RWP for review by plant workers. The main access point survey book was reviewed. The survey book was subdivided by RWP number. Many sections of the survey book were blank. Information was missing from the book to specify where the survey could be found (such as posted at the job site or on the access point bulletin board). This finding was considered important to ensure compliance with 10 CFR Part 19.12, Notice to Workers, which states "all individuals working in or frequenting any portion of a restricted area shall be kept informed of the storage, transfer, or use of radioactive materials or of radiation in such portions of the restricted area." The surveys for the reactor building were posted at the access point on a bulletin board. The surveys were clear, concise, and up to date.

Twelve RWPs were reviewed. Copies of the active RWPs were posted at the main access point. The RWPs were easy to read and provided sufficient detail to describe the radiological hazards, protective clothing requirements, and other supporting information. Several minor observations were noted about the RWPs reviewed, such as initial radiological conditions were not always listed as a range, contrary to the instructions provided in the implementing procedure.

All NRC inspector observations, including procedure typos, were presented to the licensee's representatives for potential incorporation into the RWP program. None of the observations that were identified was considered to be either a health or safety concern.

3.4 Conclusions

The licensee experienced its first positive bioassay incident during decommissioning in December 1994. The amount of radioactive material that was ingested by the individual was determined to be negligible.

The licensee's activities for final surveying of the repower area were reviewed.

The licensee's contractors had established a good RWP program. The RWPs were easy to read and understand. The survey records that were reviewed were clear, concise, and up-to-date. Several minor procedure discrepancies and implementation concerns were identified and were reported to the licensee.

4 FOLLOWUP (92701)

4.1 (Open) Deviation 50-267/9302-01: Inadequate Procedure Controls for Lifting Concrete Blocks

On May 27, 1993, while attempting to move the first pie-shaped block of concrete from the prestressed concrete reactor vessel top head, the main hoist on the reactor building crane was overloaded. At that time, the NRC concluded that the cause of the event was inadequate procedure controls, which was a deviation from commitments made to the NRC in the Decommissioning Plan. Procedural inadequacies had previously been identified as problems that contributed to at least two other incidents involving the overloading of the reactor building crane.

In the response letter to the Deviation, the licensee disputed the NRC conclusion that the event was the result of inadequate procedural controls. The licensee had determined that the event was caused by a breakdown in communications. The NRC accepted the licensee's position but concluded that procedural inadequacies contributed to the event. Corrective actions taken by the licensee included upgrading the lift work instructions. The concrete block lift evolution was subsequently completed.

On October 25, 1994, the licensee experienced a load handling incident that was the result of unsafe rigging practices (documented in NRC Inspection Report 50-267/94-08), suggesting that problems continue to exist with heavy equipment lifts. The licensee's oversight committee, in a December 15, 1994, letter, concluded that industrial safety was still a concern to the committee and that additional steps needed to be considered to improve lifting and handling procedures. During February 1995, the 345-ton core support floor was successfully lifted without incident, suggesting that the lift was well planned and executed.

This Deviation will remain open pending NRC review of the licensee's response to the oversight committee concerns.

4.2 (Closed) Violation 50-267/9402-01: Multiple Examples of Procedural Violations

During an NRC inspection conducted in March 1994, five examples of failures to comply with station procedures were identified. One example involved the failure to properly label a contaminated component, the second example involved the failure to correctly wear personnel dosimetry in contaminated areas, and the remaining three examples were associated with an event that occurred on November 5, 1993.

Corrective actions taken in response to the labelling issue included extensive revision of the labelling procedure, development of a training module, technician training, and building walkdown to identify additional examples of improper labelling (none was found). The corrective actions taken were apparently ineffective because additional examples of inadequate labelling were identified during this inspection (refer to Section 2.3 of this report).

The second issue involved the failure of site workers to wear dosimetry in the correct manner. Site personnel were wearing the dosimetry inside their protective clothing to minimize the potential for losing their dosimetry, contrary to the procedural requirements on how to don and wear dosimetry. Corrective actions taken included revision of the applicable procedure to allow plant workers to wear dosimetry inside of protective clothing.

The remaining three examples occurred during a single work activity. During work in the hot storage facility on November 5, 1993, one individual exited a contaminated area without removing all protective clothing, two individuals entered a different area without signing on to the correct RWP, and an air sample taken during the work was subsequently determined to not be representative of the worker breathing zone. This issue was the subject of a radiological occurrence report and was reviewed in detail by the licensee. Corrective actions taken in response to the event appeared appropriate for the circumstances.

4.3 (Closed) Unresolved Item 50-267/9403-02: Apparent Falsification of Records

On March 25, 1994, the licensee reported to the NRC that certain radiation survey records apparently had been falsified for selected surveys conducted between 1992 and early 1993. According to the licensee, the preliminary results of an internal investigation revealed that from September to December 1992, radiation surveys related to the release of material from the site were not documented in every case but were documented fictitiously at a later time. The documentation was generated around February 1993 from log entries and memory. Surveys were then backdated to indicate that the supporting documentation was generated at the time the material was released. In addition, in early 1993, radiation surveys related to RWPs were not consistently documented but were documented fictitiously at a later time. Personnel apparently improperly used general area surveys to take credit for RWP specific surveys.

In response to the discovery of the apparent falsification of records, corrective actions taken by the licensee and the licensee's contractors were extensive:

- A stop work order was issued between March 28 and April 12, 1994, to allow for implementation of short term corrective actions.

- The SEG staff was significantly reorganized on April 8, 1994 (the SEG organizational structure was again revised in July 1994 and December 1994).

- The Westinghouse Team initiated an independent assessment of all radiation protection activities using the Management Oversight Risk Tree (MORT) method of analysis; this assessment was in addition to the licensee's ongoing third-party investigation of harassment and intimidation issues.

- Radiological occurrence reports (SEG document) and problem reports (Public Service Company of Colorado document) were issued on the subjects.

- The contractors initiated a Radiological Improvement Program (RIP) to ensure timely completion of all proposed corrective actions.

- Management meetings were held in the NRC Region IV office on June 30 and August 4, 1994.

- About 200 procedures were revised, although training on the procedure revisions were incomplete at the time of the inspection.

In the near future, the licensee plans to perform an independent quality assurance audit of the MORT and RIP findings. This special audit will be performed by a third-party contractor and not by the licensee's quality assurance staff. In summary, the licensee's corrective actions appeared to be thorough and aggressive.

In the Fort St. Vrain Oversight Committee's December 15, 1994, letter to the licensee, the committee voiced concerns with the potential for deficiencies and inconsistencies in the radiological controls program for the final site survey of the facility by SEG, based on the problems that have occurred in the past. The committee provided three options to the licensee, including: (1) utilizing the licensee's staff to perform independent oversight of program work, (2) utilizing an independent contractor to perform redundant survey monitoring of all material to be released offsite and for the final site survey, or (3) replacing the radiation monitoring program in its entirety. The first option, utilizing licensee personnel for independent oversight, is the option that the licensee apparently will pursue.

Also, SEG assigned individuals who appear to be technically competent to perform the final survey work in an acceptable, high quality manner.

This unresolved item is being administratively closed and a different internal NRC tracking system will be used to follow up on the apparent falsification of records issues.

4.4 (Closed) Violation 50-267/9406-01: Failure to Establish a Required Surveillance Procedure

During an inspection of the Decommissioning Fire Protection Plan, the NRC discovered that a fire water makeup system surveillance procedure had not been established, implemented, and maintained, contrary to the requirements of Section 5.4.1 of the Decommissioning Technical Specifications. The licensee concluded that the cause of the violation was attributed to human error during the transition of the procedure program from the operations phase to the decommissioning phase. Corrective actions taken included developing and performing the required surveillance. The licensee subsequently revised the fire protection operability requirements to delete portions of the fire water makeup system surveillance requirements. This incident appeared to be an isolated case because no other missing fire protection surveillance procedure was identified by either the NRC or the licensee.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

T. Borst, Radiation Protection Manager
S. Chesnutt, Senior Project Assurance Engineer
M. Fisher, Program Director
J. Hak, Unit Manager

1.2 Contractor Personnel

M. Buring, Radiation Protection Operations Supervisor, SEG
C. Cummin, Rad-Waste Supervisor, SEG
B. Czajkowski, Operations/Technical Support Supervisor, SEG
B. Dyck, Licensing Engineer, Westinghouse
T. Howard, Project Director, Westinghouse
W. Hug, Site Operations Manager, MK-Ferguson
V. Likar, Technical Services Manager, Westinghouse
R. McGinley, ALARA Supervisor, SEG
D. Sexton, Technical Projects Supervisor, SEG
H. Story, Project Radiation Protection Manager, SEG

The personnel listed above attended the exit meeting. In addition to the personnel listed above, the inspector contacted other personnel during this inspection period.

2 EXIT MEETING

An exit meeting was conducted on February 9, 1995. During this meeting, the inspector reviewed the scope and findings of the report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspector.