ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-267/95-04

License: DPR-34

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

Facility Name: Fort St. Vrain Nuclear G nerating Station

Inspection At: Fort St. Vrain, Platteville, Colorado

Inspection Conducted: September 25-28, 1995

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

Accompanied By: Staff from the Low Level Waste and Decommissioning Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards:

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Approved: Charles L. Cain 10/17/95
Charles L. Cain, Chief — Date
Fuel Cycle and Decommissioning Branch

Inspection Summary

Areas Inspected: Routine, announced inspection of the final survey program and followup of several technical issues.

Results:

- The final survey program, as demonstrated by the level of documentation provided in the final survey packages, was being implemented in a sound manner (Section 1.1).
- The area classifications (affected or unaffected) documented in the final survey packages were generally found to be adequate (Section 1.1).

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- The survey maps provided in the final survey packages properly reflected the actual layout of the associated areas in the plant. The grid locations were easily identified in the plant using the maps (Section 1.1).

- Overall, the concepts of As Low As Reasonable Achievable (ALARA), sound engineering judgement, and conservatism needed to be strengthened in the final survey program, especially during survey package development (Section 1.1).

- Several program areas briefly inspected and found satisfactory included the training and quality assurance programs. Site staffing was adequate and improving although the organization chart remained out-of-date (Sections 1.3, 1.4, 1.5).

- Several technical issues were reviewed, including the discharge of radioactive material into the plant effluent pathway, hazardous material training provided to employees, the reliability of the digital dosimeter system, the licensee's response to four State of Washington violations, and followup of a NRC Inspection Followup Item (Section 2).

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

- Inspection Followup Item 50-267/9504-01 was opened (Section 1.1).

- Inspection Followup Item 50-267/9502-01 was left open (Section 2.2).

<u>Attachment</u>:

- Persons Contacted and Exit Meeting

DETAILS

1 CLOSEOUT INSPECTION AND SURVEY (83890)

Portions of the final survey program were inspected to determine if the licensee had implemented the program in compliance with the requirements established in the NRC-approved Decommissioning Plan and Final Survey Plan. In addition, representatives from the Oak Ridge Institute for Science and Education (ORISE), contractors for the NRC, performed random confirmatory surveys and side-by-side surveys to independently assess the reliability and validity of the survey methods used and results obtained by the licensee.

1.1 Final Survey Program Implementation

The process of decommissioning typically includes ceasing site operations, removing residual radioactivity, and terminating a license. In order for a license to be terminated, the remaining residual radioactivity (if any) left at the site must satisfy the release criteria which the NRC has determined to be environmentally acceptable. Following the completion of decommissioning activities, a radiological survey is performed to determine the final condition of the site (or portions of the site). The primary purpose of this final survey is to demonstrate that the release criteria established by the NRC have been met.

Final survey plans are normally developed by the licensee as part of the decommissioning process. These plans typically include: a list of the types, numbers, and locations of measurements and samples to be obtained; information on the equipment and techniques to be used; the methods to be used to interpret and evaluate the survey data; and quality control measures for ensuring the validity of the data.

The licensee's updated final survey plan, which combined the original NRC-approved final survey plan and three supplements, was submitted to the NRC on May 25, 1995. In addition, the licensee submitted proposed revisions to the criteria for classification and investigation of final radiological survey areas to the NRC on June 28, 1995. (These modifications were requested, in part, to reduce the cost of the final survey program.) The updated final survey report and proposed modification documents were approved by the NRC on July 17, 1995.

The licensee planned to perform the final survey in segments. The repower area (a 5-acre section of land in the southeastern corner of the site where gas turbines and supporting equipment were being installed) was surveyed first, followed by buildings outside of the security fence (designated by the licensee as Group A). The areas being surveyed at the time of the inspection included the smaller buildings inside the security fence but outside the power block (Group B) and portions of the effluent discharge flow path (Group E). Areas to be surveyed in the future include the turbine building, portions of the reactor building, and finally, the prestressed concrete reactor vessel area.

To perform the final survey, the plant was subdivided into discrete sections. The licensee used the term "survey unit" to describe each contiguous area with similar characteristics and contamination potential. A "survey package" was prepared for each survey unit. Survey packages were collections of information in a standardized format for controlling and documenting field measurements taken during the final survey. The survey package typically included survey instructions, a grid map, survey measurement data sheets, and a history of the area.

The history files were a compilation of information prepared for use in planning the survey; it summarized the operational history, characterization survey data, operational surveys, and other information as appropriate. In general, the history file was used to support a designation of each survey unit as either affected (areas having potential or known radioactive materials or contamination) or unaffected (areas not expected to contain radioactive materials or contamination).

Once a final survey had been started in an area, the area was supposed to be segregated by appropriate boundaries and markings to isolate that area from the remainder of the plant. These isolation and control measures were implemented to prevent the potential spread of radioactive contamination into areas verified free of contamination.

During the inspection, seven survey packages were thoroughly reviewed. In addition, the history files of six more survey packages were reviewed to determine if the packages were properly classified. Also, portions of three survey packages were walked down in the plant to determine if the grid maps and locations marked on the maps were accurate and properly reflected actual field conditions, and to determine if isolation and control measures were properly implemented.

In general, the survey program, as documented in the survey packages, was being implemented in a sound manner. The packages were noted to be thorough although the survey packages were incomplete at the time of the inspection. For example, the data reduction information (the analysis) was not available in the packages. In addition, several minor inconsistencies were noted.

The area classifications were found to be adequate, with one exception. One survey package (designated as B005) was noted to be classified as "unaffected" although the history file indicated that radioactive material had been stored in this area of the plant. The NRC concluded that this area should have been designated as "affected" in the survey package.

In addition, a potential weakness in classifications was noted involving areas being classified as "unaffected" that were adjacent to areas classified as "affected." In these cases, the licensee needed to ensure that the justifications for an unaffected classification were sound, based on good engineering judgement, and well documented.

Overall, the NRC concluded that the concepts of As Low As Reasonable Achievable (ALARA), sound engineering judgement, and conservatism needed to be strengthened in the final survey program, especially during survey package development. For example, the history file for survey unit B005 indicated that contamination had been stored in the area in the past, a hot soil laboratory was previously located in the area, and a contaminated area was found in this unaffected area during the final survey. The investigation that followed the discovery of the positive survey findings appeared to be limited in nature. Besides being reclassified as "affected," additional radioactive survey scans may have been warranted in this area. The minimum level of action appeared to have been taken by the licensee when ALARA principles and sound engineering judgement appeared to be more appropriate for the circumstances.

A field walkdown of three survey packages was performed. Overall, the survey maps properly reflected the actual layout of these areas in the plant. The grid locations were easy to identify in the plant with the use of the maps. Several minor problem areas were noted. For example, several labels were missing, primarily on floors, because the adhesive had worn off or because the labels were destroyed by work activities in progress in the areas.

Isolation and control measures were properly implemented in each area except one. The Chicago Pneumatic Building No. 28 did not have the required green isolation label visible from the exterior of the building. This label was located on the exterior of the building door, but the label was not visible because the door had been propped open. This finding was noted to be representative of a generic problem that was faced by the licensee, the labelling of doors frequently left open. Labelling both sides of the door may be inappropriate because the areas on both sides may not have been finally surveyed. This problem area appeared to warrant additional licensee attention to ensure that proper isolation and control measures are being maintained. No radioactive material was identified in any of the areas inspected, suggesting that the isolation and control measures were generally successful.

Prior to the inspection, a proposed checklist and inspection plan were developed by the NRC. All items on the checklist were not completed, in part, because the program areas were not fully implemented. The inspection items not fully reviewed included:

- The analysis of survey results (data reduction activities).

- The review of completed survey packages to ensure compliance with the program requirements.

- The implementation of the commitments made in the June 28, 1995, letter to the NRC with respect to the classification and investigation criteria.

- A review of the instrument sensitivities.

- The implementation of the final survey quality control activities.

- A review of records management, including the control of computer databases of the final survey data, the quality control oversight of these databases, and development of the final survey quality control procedure "testing of computer calculations."

- Ensuring that each survey package was a stand-alone, auditable document.

These areas will be reviewed during a future NRC inspection (NRC Inspection Followup Item 50-267/9504-01).

1.2 ORISE Site Visit

During the inspection period, two members from ORISE visited the facility to perform an instrument comparison check and to perform random confirmatory surveys of selected areas of the plant.

The instrument comparison check consisted of side-by-side field measurements utilizing ORISE's radiological survey instruments and Scientific Ecology Group's (SEG) survey instruments. (SEG, part of the Westinghouse Team performing the decommissioning of the site, was contracted to perform the final survey for the licensee.) Three side-by-side measurements were taken at three different locations to compare the radioactivity levels measured by each set of instruments at each location.

The first comparison check was performed on the Electrical Building No. 14 floor, an area considered to be unaffected. Both groups used gas flow proportional detectors to measure the radioactivity of the uncontaminated concrete. A second side-by-side check was performed in the Battery Room on Level 5 of the Turbine Building, also an unaffected area. The third check was performed in the Resin Changeout Area, an affected area located on Level 3 in the Reactor Building. In addition, independent survey scans were performed in three survey units, including the Training Building, for comparison of ORISE's as-found measurements with the results provided in the survey packages for these survey units.

To ensure that the comparison checks were completely random and not planned, the survey technician and the instrument used by the technician were randomly chosen by the NRC. A calibration history review was performed on the randomly selected instrument. The instrument's calibration was noted to be up-to-date. In addition, the technician's training records were reviewed, and the technician was noted to be properly trained and authorized to perform final survey work.

The results of the ORISE measurements were not available at the end of the inspection period; therefore, no conclusion was drawn from this portion of the inspection.

1.3 Final Radiation Survey Program Staffing

The staffing levels of the onsite SEG organization were reviewed. During the previous NRC inspection, several key positions in the organization were unfilled or were being performed on a part-time basis by other individuals in the organization. Since that time, the licensee has added staff members to the final survey program but was not finished with the staffing effort. The SEG staff currently consisted of 100 people, up significantly since the last inspection.

The inspector noted that the SEG organization chart was still out of date. The last approved organization chart was dated December 1994. SEG management was aware of this weakness and was actively attempting to update their organization chart.

1.4 Quality Assurance Audits

Technical Specifications 5.3.6 states that audits of decommissioning activities shall be performed under the cognizance of the decommissioning safety review committee. The licensee's quality assurance department performed an audit of the radiation protection program, including the final survey program, in June-August 1995. The licensee's audit of the final survey program was briefly reviewed during the inspection. Overall, the quality assurance audit appeared to be a thorough assessment of the radiation protection program, including the final survey program.

Final survey program strengths were identified during the audit including the staffing of key positions with competent individuals and management commitment to the objectives of the final survey. Weaknesses were identified with the conflicting procedure requirements involving the final survey organizational structure, the reporting relationships and responsibilities for several key positions, standardization of position descriptions, and licensee oversight of the final survey staffing levels.

In addition, the decommissioning contractor performed quality assurance audits that were independent of the licensee's audit. The Westinghouse Team quality assurance department performed an audit of final survey activities in August 1995, while the SEG quality assurance department performed an audit in March 1995. The contractor's audits were not reviewed during this inspection.

1.5 Training Program

The licensee's program for training individuals to perform final survey work was briefly reviewed. The training program had been established and consisted, in part, of job qualification standards and lesson plans. A job qualification card matrix was developed to keep track of individuals versus training completed. No individual associated with the final survey was identified that was unqualified to perform the tasks required for the job.

1.6 Embedded Piping

The embedded piping issues were discussed with the licensee during the inspection. The site has about 30,000 feet of "affected" embedded piping, including about 22,000 feet of 1-inch diameter piping embedded with elbows. Surveying and decontaminating this piping will be a potentially challenging endeavor on the part of the licensee. No conclusions were drawn during the discussions with the licensee. The licensee committed to submit a technical document to the NRC in the near future which will address their plan of action for handling embedded piping.

During this inspection, tours of the Reactor Building were performed. Representative examples of embedded piping were pointed out to the inspectors during the tours.

1.7 Conclusions

The survey program, as demonstrated by the level of documentation provided in the final survey packages, was being implemented in a sound manner. The survey packages were noted to be thorough documents although the survey packages were incomplete at the time of the inspection.

The area classifications that were documented in the survey packages were found to be adequate, with one exception. In addition, a potential weakness in classifications was noted involving areas being classified as "unaffected" that were adjacent to areas classified as "affected." The survey maps provided in the survey packages properly reflected the actual layout of these areas in the plant. The grid locations were easy to identify in the plant. Several minor problem areas were noted, including the use of boundary markers on doors frequently left open.

Overall, the NRC concluded that the concepts of As Low As Reasonable Achievable (ALARA), sound engineering judgement, and conservatism needed to be strengthened in the final survey program, especially during survey package development.

Several program areas briefly inspected and found satisfactory included the training and quality assurance programs. A conclusion of the effectiveness of the final survey quality control program was not made. Site staffing was adequate and improving although the organization chart remained out of date.

A number of final survey program areas were not inspected and will be reviewed during a future NRC inspection.

2 FOLLOWUP (92701)

2.1 Followup of Previously Identified Technical Issues

During the inspection, several technical issues, previously identified by the NRC as requiring a followup, were reviewed to determine if the issues had been acceptably resolved by the licensee.

2.1.1 Discharge of Radioactive Materials into the Plant Effluent Pathway

Radioactive material was released into the environment from the site in both gaseous and liquid form. The gaseous form was released via a ventilation stack while the liquid form was released in batches via the effluent pathway. Plant liquid effluent was normally discharged from the site into the environment through (in order) the Goosequill Ditch, Jay Thomas Ditch, and Farm Pond. The Goosequill Ditch is a concrete-lined ditch that is about 6000 feet long, 3-feet wide and 3-feet deep. The Jay Thomas Ditch is about 2300 feet long, 3- to 6-feet wide and 3-feet deep. Following the Jay Thomas Ditch, the effluent flows into the 25-acre Farm Pond.

Just prior to a liquid release, grab samples were taken to quantify the amount of radioactive material in the liquid. The allowed release rate was then calculated per guidance provided in the Offsite Dose Calculation Manual. Radioactive material was normally detected in the liquid prior to release; however, the effluent was diluted to ensure that the release limits established in 10 CFR 20, Appendix B, were not exceeded.

Samples of the effluent pathway have been taken and analyzed for its radioactive constituents as part of the environmental monitoring and site characterization programs. SEG performed site characterization sampling of the effluent pathway in late 1994 and early 1995. Roughly 700 samples were taken at various locations along the effluent pathway, including sediment, surface, subsurface, and vegetative samples. Radionuclides identified in the samples included cobalt, cesium, iron, strontium, and europium in varying amounts.

The sample results indicate that some remediation may be necessary, but the amount of radioactive material identified was not significant. Conceptual modelling performed by SEG revealed that the dose to individuals, using conservatism and considering all possible credible pathways, indicated a maximum dose equivalent of 39 millirems per year from radionuclides in the effluent pathway (worst case scenario).

SEG did not attempt to locate tritium in the environment because they were trying to assess the impact of selected effluents on the pathway following the completion of unique decommissioning activities (such as the release of the shield water system water volume).

Environmental monitoring was performed to comply with license requirements. As part of the environmental monitoring program, surface water samples from the effluent pathway were taken and analyzed to determine the tritium concentrations in the water. Elevated levels of tritium were noted in the third and fourth quarters of 1994. The highest observed reading was 2700 picocuries per liter; the EPA drinking water limit is 20,000 picocuries per liter. The annual report noted that the level of tritium measured in 1994 was lower than levels identified during the years that the plant was operational.

In addition, cesium-137 was identified in the effluent pathway's surface water; however, the amount of cesium in the effluent pathway water was statistically similar to the amount found in the water upstream of the site. This suggested that the source of the cesium was not the plant but was the result of atmospheric fallout. Other radionuclides were found in the surface water at very low levels during environmental monitoring. The licensee suspected that these findings were indicative of false positive readings.

Surface sediment and fish samples were also taken in the effluent pathway. Low levels of cesium were noted in the sediment samples while cesium and cobalt were noted in the fish samples in small amounts.

In summary, radioactive material was present in the effluent pathway; however, the levels identified were not significant and would not pose undue risks to individuals working in or around the ditch for short periods of time. However, sampling results suggest that sections of the effluent pathway may require remediation as part of the site decommissioning process.

2.1.2 Hazardous Material Training of Personnel

The amount of personnel safety training provided to site employees was reviewed. Hazardous material training was provided to all plant workers during general employee training. Site access training included Occupational Safety and Health Administration (OSHA) hazard communication standards, the law that gives employees the right to know the hazards of the material with which they come in contact with. This training also included a video on the subject.

In addition, site radiological hazards were also discussed during radiation worker training. This training included discussions of 10 CFR 19, "Notices, Instructions and Reports to Workers, Inspections and Investigations" and 10 CFR 20, "Standards for Protection Against Radiation." Finally, both radiological and industrial airborne hazards were discussed in the respiratory protection training classes.

Site postings were inspected. OSHA Form 2203, "Job Safety and Health Protection," and the new NRC Form 3, "Notice to Employees," were prominently displayed onsite in multiple locations.

In summary, site workers were given opportunity to understand their rights and responsibilities associated with radiological and industrial hazards because these subjects were discussed in site access training classes. Also, OSHA and NRC-required postings were displayed on the site's bulletin boards. These postings listed worker's rights and responsibilities, as well as telephone numbers workers could use for voicing questions or concerns to these government agencies.

2.1.3 Reliability of the Digital Dosimeters

External radiation dose monitoring is accomplished through the use of thermoluminescent dosimeters (TLD) and self-reading pocket dosimeters (SRD) or digital alarming dosimeters (DAD). The official record of external dose is normally obtained from the TLDs. The SRDs or DADs are used to track doses between TLD processing periods and are used as backups to the TLDs.

During normal entries into and exits from the radiologically restricted areas, workers usually electronically signed on to the applicable radiation work permit via the digital dosimeter system. The digital dosimeter system consisted of a FASTRAK computer, four DAD readers, a converter located between the computer and readers, and roughly 60 DADs. Whenever the digital dosimeter system was out of service or DADs were unavailable, the SRDs were used and workers were manually signed into and out of the restricted area.

The DADs have experienced problems in the past, including an under-response of about 20 percent with respect to the TLD readings (the digital dosimeter system was providing dose values that varied by about 20 percent below the values obtained by TLDs), the failure of wires on the DAD floating detector board, and blank screens because of bad displays or batteries. Corrective actions taken included recalibration of the dosimeters to compensate for the under-response and rewiring the floating detector boards. Other ongoing problems with the DADs included lack of spare parts and costs of repairs.

The licensee changed TLD vendors in January 1995 and changed the frequency of TLD processing from monthly to quarterly in April 1995. The licensee then experienced a new problem; the DADs apparently were over-responding when compared to the results of the new TLDs.

In reaction to this new problem, the licensee performed an investigation that included the controlled exposure of a phantom with TLDs, SRDs, and DADs. The results of these investigations suggested that the TLDs were under-responding (a negative bias) to a controlled exposure while the DADs were over-responding. The results also indicated that the SRDs were responding properly without an over- or under-response.

The licensee concluded that the site TLDs' under-response were acceptable because the minor variations were within the limits established for laboratory accreditation. Corrective actions being considered by the licensee included readjusting the DADs to respond more consistently with the new TLDs.

In summary, licensee action to ensure accurate dose monitoring was appropriate. The TLD results were the official exposure values assigned to individuals (except during unusual situations, such as incidents involving lost TLDs).

2.1.4 Violations of State of Washington Administrative Codes

Since the beginning of the decommissioning process, 332 radioactive waste shipments have been made from the site. Between January and August 1995, 71 offsite shipments were made, mostly to a commercial low-level waste disposal site near Richland, Washington. Four State of Washington violations were recently issued to the licensee because of noncompliance with the State's Administrative Codes related to shipping of material for burial. During this inspection, the four violations, and corrective actions taken in response to the violations, were reviewed to independently ascertain whether the four events were related and whether corrective actions taken were appropriate for the circumstances.

On April 3, 1995, a shipment of radioactive waste material was received at the waste disposal site near Richland. State inspection of the shipment revealed that nine barrels had as-found survey readings that were different than the readings listed on the shipping manifest. The as-found readings were roughly off by a factor of ten from the manifest readings. Although the root cause of the incident was not clearly identified, the most probable causes included survey instrument malfunction or the use of the incorrect scale on the survey instrument during the initial survey of the barrels.

About 3 weeks prior to the shipment, the survey instrument used during the initial survey malfunctioned and was subsequently repaired. Following the discovery of the barrel survey reading discrepancies, a calibration check and functional test were performed on the suspect instrument. The instrument passed the calibration check and functional test without a failure.

Regardless, corrective actions were taken by the licensee which included programmatic changes on how to survey waste shipments prior to departure from the site for disposal. For example, each shipment now receives an independent verification survey (two technicians using two separate survey instruments). Also, the audible feature of the survey instrument was required to be used during the surveys because the audible function was not affected by a change in instrument scales.

On April 10, 1995, the State noted that a box had arrived at the disposal site with an as-found survey reading of 120 millirems per hour while the reading listed on the manifest was 30 millirems per hour. The licensee concluded that the load apparently shifted in transit. Filler material used to fill void spaces in the container apparently had settled during the shipment of the container. Corrective actions taken included modifying the method used to fill shipping containers.

The third violation involved the inadequate marking of eight drums. On June 30, 1995, the burial site received a shipment that contained drums that were not clearly marked with the appropriate labels. The drums were marked on the sides but when the barrels were stacked adjacent to each other the markings could not be seen from the top. This method of marking was used, in part, to ensure that the container was labelled in case the container lid came off following a transportation incident. Corrective actions taken included changing the method of marking containers to include markings on the lids of barrels, and ensuring that each container was clearly marked and that the markings were visible prior to shipment.

On August 4, 1995, the burial site received a shipment that contained two boxes and one drum with as-found survey readings that were different from the readings listed on the shipping manifest. The apparent cause of the drum survey reading discrepancy was the failure to fill the drum to the 100 percent full level with filler material which allowed the contents of the drum to shift during shipment. The apparent cause of the box survey reading discrepancies were a malfunction of the computer program that generated the data listed in the manifest. Corrective actions taken included changing the shipping program to increase the filling of containers with fill material to the 100 percent level (the 90 percent level had been the previous benchmark) and performing a thorough review of the manifest prior to shipment.

In response to the recent State violations, the licensee sent a letter to the decommissioning contractor which requested that the contractor perform a review of the incidents and address the potential programmatic issues that were involved. The licensee's concerns were valid because the State could deny the licensee access to the disposal site if the nature and quantity of violations were determined to be unacceptable to the State.

The corrective actions taken by the licensee for these four incidents appeared appropriate for the circumstances. However, the corrective actions were not explicitly being incorporated into the licensee's or contractor's radioactive shipment program or implementing procedures.

For example, one corrective action commitment was to perform independent verification surveys of materials being shipped. The procedure that implemented this action simply stated to "perform a contamination and radiation survey of the exterior of the containers." Although the licensee's representatives were performing the required verification surveys, the failure to revise the procedure to clearly indicate that a verification survey was required may result in the licensee or contractor "forgetting" the commitment made to the State, resulting in a noncompliance with the commitments.

A second example involved the filling of containers to the 100 percent full level prior to shipment. The implementing procedure stated that the containers were required to be filled to a level greater than 85 percent full. Again, the failure to revise the implementing procedure to clearly state that the containers were to be filled to the 100 percent level may result in future noncompliance with State commitments or repeat violations.

2.2 (Open) Inspection Followup Item 50-267/9502-01: Review of the Third-Party Independent Verification Program

Section 4.7 of the Decommissioning Plan states that a third-party independent verification of the final survey will be performed as an audit of the final survey plan. This independent verification would include selected measurements, sampling, and analysis as required to confirm the validity of the final survey. In addition, this independent verification program was to be developed with a structure similar to the final survey plan.

During an inspection conducted in March 1995, the licensee did not have copies of the third-party's program, plan, or procedures available for review. An Inspection Followup Item was created to ensure that the NRC reviewed the program.

During this inspection, selected implementing procedures were available and were briefly reviewed for consistency with the NRC-approved final survey plan. However, the third-party's administrative program documentation was not available for review (a "proposal" of the program was available for review). In addition, the results from the third-party's final survey of the repower area (performed in early 1995) were still not available for review.

The NRC inspectors questioned whether the third party was capable of performing "quality-related" work because the third party had not been inspected by the licensee's quality assurance organization. The licensee could not clearly indicate whether the third party was or was not subject to oversight by their quality assurance organization. Technical Specifications 5.3.6.b states that audits of decommissioning activities shall be performed under the cognizance of the decommissioning safety review committee, including any area of facility activities considered appropriate by the committee.

This Inspection Followup Item will remain open pending the decommissioning safety review committee's decision on whether the third party was subject to a quality assurance audit or whether the third party was exempt from the audit requirements, and pending NRC review of the remainder of the third-party's program to ensure compliance with the Decommissioning Plan.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

T. Borst, Radiation Protection Manager
S. Chesnutt, Senior Project Assurance Engineer
M. Fisher, Decommissioning Program Director
M. Holmes, Project Assurance Manager
D. Seymour, Senior Quality Assurance Engineer

1.2 Contractor Personnel

D. Blain, Field Operations Coordinator, SEG
M. Buring, Radiation Protection Operations Supervisor, SEG
B. Dyck, Licensing Engineer, Westinghouse
T. Howard, Project Director, Westinghouse
W. Hug, Operations Manager, MK-Ferguson
M. Lauer, Senior Radiological Engineer, SEG
V. Likar, Technical Services Manager, Westinghouse
B. Mann, PSC Project Assurance Consultant
R. McGinley, ALARA Supervisor, SEG
D. Parsons, Radiological Engineer, SEG
J. Rood, Final Survey Lead Engineer, SEG
D. Schult, Technical Oversight, SEG
D. Sexton, Technical Support Supervisor, SEG
H. Story, Project Radiation Protection Manager, SEG
M. Zachary, Final Survey Operations Supervisor, SEG

1.3 Oak Ridge Institute for Science and Education

E. Abelquist, Project Leader, Environmental Survey and Site Assessment Program
R. Morton, Health Physics Technician

1.4 NRC, Office of Nuclear Materials Safety and Safeguards

L. Bell, Section Leader, Division of Waste Management
C. Pittiglio, Senior Project Manager, Division of Waste Management
D. Fauver, Senior Project Manager, Division of Waste Management

1.5 NRC, Region IV

R. Evans, Health Physicist, Division of Radiation Safety and Safeguards

The personnel listed above attended the exit meetings, except for the ORISE representatives. In addition to the personnel listed above, the inspectors contacted other members of the site staff during this inspection period.

2 EXIT MEETING

Two exit meetings were conducted. An exit meeting for the final survey program inspection was conducted on September 27, 1995. An exit meeting for the followup of several previously identified technical issues was held on

September 28, 1995. During these meetings, the inspectors reviewed the scope and findings of the report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspector.

During the first exit meeting, the licensee was requested to submit the results of their side-by-side survey measurements, performed simultaneously with the ORISE representatives, to the NRC in an expeditious manner. In addition, the licensee was requested to submit instrument correlation data to the NRC in a timely manner. The NRC planned to use this information, in part, to assess the accuracy, reliability, and validity of the licensee's final survey program.