ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-267/96-01

License: DPR-34

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

Facility Name: Fort St. Vrain Nuclear Generating Station (FSV)

Inspection At: Fort St. Vrain, Platteville, Colorado

Inspection Conducted: January 22-25, 1996

Inspectors: L. C. Carson II, Health Physicist
R. J. Evans, Health Physicist

Accompanied By: M. F. Weber, Chief
Low Level Waste and Decommissioning Projects Branch (LLDP)
Office of Nuclear Materials Safety and Safeguards (NMSS)

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Approved: D. Blair Spitzberg, Ph. D., Chief — 3-5-96 Date
Nuclear Materials Licensing Branch

Inspection Summary

Areas Inspected: Routine, announced team inspection was conducted to assess the licensee's implementation of the Final Survey Plan. This assessment included collecting independent measurements, evaluating radiation instrumentation usage, analyzing data reduction techniques, examining the records management of survey packages, reviewing the quality assurance/quality control program, and closing out the followup item identified in the September 1995 inspection.

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Results:

- An unresolved item was identified concerning variations in shielded local area background measurements (Section 2.1).

- Overall, the licensee had developed and implemented a state-of-the-art program for data collection and analysis. The absence of a routine quality control program for validation of computer software was identified as a potential area for improvement. Several procedures had not been updated to reflect the latest methods for inputing data and selecting material background values (Section 2.3).

- Generally, the survey program and survey packages were thorough and documented adequately. Survey package investigations and reclassification were adequate (Section 2.4).

- Several technical issues were identified during the performance of independent measurements and reviews of final survey packages. The inspectors identified the following concerns: (1) whether there is a bias in instrumentation response which overestimates the amount of contamination present (Section 2.1.1); (2) whether corrections for hard to detect nuclides (HTDNs) should be made to survey results in unaffected areas that exceed 25 percent of the surface contamination limits; (3) whether investigations of suspect measurements were adequately conducted to justify removing the original measurement from the survey database (Section 2.4.5); and (4) whether the scan survey coverage percentage in nonsuspect affected areas should be increased (Section 2.4).

- The use of final survey isolation and control boundaries to control areas that had been final-surveyed was improved in the reactor building (Section 2.6).

Summary of Inspection Findings:

- Unresolved Item 267/9601-01 was opened (Section 2.1)
- Inspection Followup Item 267/9504-01 was closed (Section 5.1).

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

On January 22, 1996, the licensee met with Region IV staff to present an overview of the history, status, and future plans for the Fort Saint Vrain Site. The licensee also discussed site repowering status, corporate merger plans, lessons learned, regulatory concerns, and transfer of the spent fuel and storage installation to the Department of Energy.

The major tasks in progress at Fort Saint Vrain were dismantling and decontamination of the reactor building embedded piping, decontamination of secondary systems, turbine building surveys, and implementation of the Final Survey Plan. Decommissioning work was being performed by the Westinghouse Team, which consisted of personnel from Westinghouse Electric, Scientific Ecology Group, and Morrison Knudsen-Ferguson. Only the Repower Area at the Fort Saint Vrain site had been decommissioned and released for unrestricted use. The Repower Area was a section of land in the southeastern corner of the site that was released to support the installation of gas turbine power units in May 1995.

Decommissioning includes ceasing nuclear operations, decontamination, removing residual radioactivity, and terminating the license. In order for a license to be terminated, the amount of residual radioactivity left on site must meet NRC release criteria. Following the completion of decontamination activities in a specific area, radiological surveys are performed according to the NRC-approved Final Survey Plan. The primary purpose of the Final Survey Plan is to demonstrate that the release criteria established by the NRC have been met.

2 CLOSEOUT INSPECTION AND SURVEY (83890)

The NRC team assessed the licensee's compliance with the NRC-approved Decommissioning Plan and Final Survey Plan. Representatives from the Oak Ridge Institute for Science and Education (ORISE), an NRC contractor, performed independent side-by-side and scan surveys with the licensee. ORISE's survey results were used to assess the reliability and validity of the survey methods, instruments, and results obtained by the licensee.

A Final Survey Plan was developed by the licensee as part of the Decommissioning Plan. The plan included 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.

This inspection covered the implementation of the Final Survey Plan in the following areas: independent measurements, instrumentation usage, data reduction techniques, records management of survey packages, quality

assurance/quality control, and followup of issues identified during the September 1995 inspection.

2.1 Independent Measurements

During the inspection, an ORISE team and two licensee contractor teams performed instrument comparison checks and confirmatory surveys of selected areas of the facility. Licensee survey technicians and licensee instruments were randomly chosen by the NRC. Similar gas flow proportional detectors were used to measure radioactivity by the teams. Calibration history reviews were performed on the selected instruments and were found to be up-to-date.

2.1.1 Direct Measurement Comparisons

Instrument comparisons consisted of side-by-side direct measurements of surface radioactivity by ORISE personnel using ORISE radiological survey instruments and measurements made by two teams from the Scientific Ecology Group (SEG) using SEG survey instruments. SEG was contracted by the licensee to perform the final surveys.

The licensee's direct measurement results were statistically higher than ORISE's results when radioactivity was present, and at area background levels, no differences existed. Inspectors found one potential explanation for the licensee's results being higher. The licensee calibrated detectors with the probe about 0.27 cm from the radiation source, while ORISE calibrated detectors with the source about 0.10 cm from the probe. This calibration method may result in the licensee using a lower efficiency than ORISE. This bias in the licensee's results appeared to overestimate the amount of contamination present. Inspectors recommended that the cause and amount of the bias be determined by the licensee.

Side-by-side direct measurements were taken at four different locations to compare the radioactivity levels measured by each set of instruments. The first side-by-side measurements were performed on the floor of the battery room in the turbine building, an area considered to be an unaffected area. A second side-by-side check was performed in the lube oil storage room of the turbine building, also an unaffected area. The third and fourth side-by-side checks were performed in the barrel evaporator room, an affected area located on Level 9 in the reactor building. Radioactivity in the affected areas measured between 2000 and 4000 disintegrations per minute per 100 centimeters squared (dpm/100cm^2); 50 percent and 100 percent of the hard to detect nuclide (HTDN) corrected guideline value, respectively. ORISE and SEG collected ten measurements in each area at a single point. Additionally, ORISE and SEG measured local area background radiation in the vicinity of the single point measurements.

Reviews of side-by-side measurements revealed a variance in the local area background radiation data obtained by the two SEG teams. The affected area background measurements at 50 percent of the HTDN corrected guideline value of 2000 dpm/100cm^2 varied an average of 171 dpm between the two SEG's teams.

During the ORISE direct measurement comparison with the licensee, inspectors questioned the accuracy of the licensee's method for determining background. Inspectors were concerned that the licensee's method may be overestimating background, thereby underestimating the final survey results. The observed variations in the local area background measurements could affect the validity of some survey results. In addition, the measurement variations could affect licensee decisions to investigate elevated radioactivity measurements. For example, if the licensee's investigation limit was 3000 dpm/100cm^2, a local area background correction that was biased high could cause a net measurement result to be reported at less than 3000 dpm/100cm^2.

The basic equation the licensee used to correct unshielded measurements, also called gross measurements, was as follows:

> Net Result (dpm/100cm^2) = Unshielded Measurements (Gross) - Shielded Measurements (Local Area Background) - Natural Material Background

The licensee evaluated background surface activity for a given material by dividing the background into two components: natural material (e.g., concrete) background and the local area exposure rate background. The licensee had collected a series of shielded counts (using a 300 mg/cm^2 plexiglass beta shield) at a selected background location that represented the local area exposure rate contribution to the detector response. Shielded measurements were followed by unshielded measurements that included the contribution from both the local area exposure rate and the natural material background. The difference of these two values yielded the natural material background for a specific material (i.e., concrete, steel, wood, etc.). This natural material background, plus the specific local area exposure rate contribution to the detector were subtracted from the gross surface activity measurements. During the measurement comparison between ORISE and the licensee, the shielded background varied between the two SEG teams in the barrel evaporator room. ORISE also experienced an increase in the shielded background when the detector was moved from the 2000 to 4000 dpm/100 cm^2 surface activity location.

Licensee representatives did not have a written explanation on how local area background and plexiglass shielded survey data variation affected final survey package results. According to 10 CFR Part 20, the licensee is required to make and evaluate surveys that are reasonable under the circumstances. The inspectors determined that the licensee's method for determining background needed further testing to ensure that background values were not being overestimated. This matter will be considered an Unresolved Item (267/9601-01). An unresolved item is matter which requires additional information to ascertain whether the issue in question is an acceptable item, a deviation, a nonconformance, or a violation. In order to resolve this item, the NRC needs

to review information provided by the licensee that demonstrates the following:

- That background values applied do not increase as the radioactivity level increases, resulting in an underestimate of net survey results.

- That background measurements collected by different licensee technicians using instruments with different detector shields were consistent.

2.1.2 Scan Surveys

ORISE collected scan survey data at the following five survey units locations:

- C005: Turbine Building Level 5 South General Area
- C031: Turbine Building Level 7 Turbine Deck
- C046: Building 10 Ground Floor
- F044: Level 3 East Reactor Building Resin Changeout Room
- F045: 1B/1D Cooling Water Heat Exchange Room

A "survey unit" was a contiguous area with similar characteristics and contamination potential where at least 30 measurement locations exist. Scientific Ecology Group (SEG) had previously collected scan survey data in three of the above survey units. Scan surveys were qualitative radiation measurements conducted by moving the detector probe at a constant rate of motion and maintaining the probe at least one centimeter from the surface of the survey point. This method was used by the licensee to quickly characterize an area. If elevated radiation levels were identified, an investigation would be conducted for the given location. Generally, ORISE's scan surveys were consistent with the licensee's. However, ORISE identified a 2000 dpm/100cm^2 spot that SEG missed and, likewise, SEG identified a 3600 dpm/100cm^2 spot that ORISE missed. This was not significant based on the licensee's release criteria. Inspectors were satisfied with the results of the scan comparisons.

2.2 Survey Instrument Usage, Sensitivities, and Measurement Techniques

Inspectors reviewed several aspects of the licensee's radiation instrument and measurement systems to determine if the licensee was maintaining instrument measurement capabilities as specified in the Final Survey Plan.

2.2.1 Embedded Piping Measurements

Tours of the reactor building were conducted to observe examples of embedded piping. Embedded piping issues were discussed, and licensee staff demonstrated various survey methods. Embedded piping measurements involved 30,000 feet of "affected" piping, which includes 22,000 feet of 1-inch diameter piping with bends. Several different surveying and decontaminating methods were being used for embedded piping. Hydrolazing, beadblasting, and pipe grouting were some of the decontamination methods being used. Inspectors

observed mock-ups of small Geiger-Mueller detectors, small cylinder gas flow detectors, and thermoluminescent detector strings being tested for embedded piping measurements. No conclusions were drawn from the observations or during the discussions with the licensee relative to embedded piping issues. However, NRC project management indicated that the results of a technical review of the licensee's January 18, 1996, "Response to NRC Questions Regarding PSCo's Proposed Revisions to the Final Survey Plan, Involving Survey of Piping Systems and Suspect Affected Survey Units," would be submitted in February 1996. This NRC review will further address the plan of action for handling embedded piping.

2.2.2 Gamma Spectrometry for Exposure Rate Compliance

Fort Saint Vrain has significant variances in background exposure rates due to licensed and natural radioactive materials. The licensee was continuing to develop a high-purity germanium gamma spectrometry system as a measurement system that could differentiate licensed material from naturally occurring background radiation. The licensee was also developing a sodium iodide gamma spectrometry system as a cross check to the high-purity gamma spectrometry system. The license planned to have procedures in place by the end of February 1996 as part of the exposure rate determination program. Inspectors observed both the high-purity and sodium-iodide (NaI) gamma spectrometry systems.

2.2.3 Exposure Rate Correction: Pressurized Ion Chamber to Micro-R-Meter

Section 5.1.3 of the Final Survey Plan, "Exposure Rate Measurements," stated, in part, that a correction coefficient will be calculated and applied to each sodium iodide (NaI) micro-R-meter to correct the reading to the pressurized ion chamber (PIC) value. Inspectors reviewed completed final survey packages, Technical Basis Document FSV-FRS-TBD-206, Revision 0, "Background Determination" and Implementing Procedure FSV-SC-FRS-I-101, Revision 0, "Background Level Determination for Final Survey." Inspectors found that the licensee has yet to establish a correction coefficient between PICs and NaI detectors. Inspectors expressed concern because final survey packages could not be considered 100 percent complete until this matter was addressed. Inspectors explained that the implementation of the background PIC to NaI detector coefficient would have to be inspected in the future, which could result in delaying the final report reviews. The licensee indicated their intent to submit the Technical Basis Document for the PIC to NaI correction coefficient by the end of February 1996.

2.2.4 Large Area Probes

According to the Final Survey Plan, the licensee should develop procedures and calculations before using large area radiation detection probes. The NRC reviewed the use of survey instruments with a probe size greater than 125 cm^2 (large area probes). Table 4.1 of the final survey plan lists a large area probe as an instrument that may be used in the final survey. The use of large area probes were reviewed since these probes may underestimate the surface

contamination in some instances. Inspectors found that no operating procedures were in place for the use of these probes. Licensee staff informed the NRC that large area probes were not being used for the final survey and are not planned to be used in the future. Inspectors had no concerns with this decision.

2.3 Data Reduction

The licensee's methods of analyzing the results of the final survey (data reduction activities) were reviewed to ensure that an effective process had been developed and implemented. This evaluation included the methods used by the licensee to compile, store, transfer, analyze, and document the data collected during the final survey. Quality control management of the data reduction activities was also inspected. Surface activity, soil activity, and exposure rate measurements were taken by the licensee. Because of the large number of data, reliable methods of data collection and processing had to be developed and implemented. These data reduction methods also ensured that radiological measurements were valid and accurate. The results of the NRC's review are discussed further below.

2.3.1 Data Reduction Activities Program Review

Inspectors reviewed the licensee's method of obtaining and processing surface activity measurements. Generally, microprocessor-based survey instruments were used to collect surface activity readings at predetermined locations. Following the completion of the data collection, the data stored in the instruments was "downloaded" into a computer database. Decommissioning Survey Reports generated from the database, contained information such as the time and date of the survey, location of each survey, and the activity or exposure rate reading at each survey location. These reports were subsequently added to the applicable survey package.

Following download into the computer, the survey data was processed using the "DBACORR" computer program. DBACORR was an internally developed program that generated release records for survey units and cover sheets for survey areas and units. Survey areas were subdivided into survey units, and survey packages were being developed for each survey unit. A release record was a document compiled for each survey unit which demonstrated that the unit was suitable for unrestricted use. Release records contained the evaluation of the survey data and supporting information, such as statistical analysis. Release records provided a concise record of the survey results and the basis for a conclusion that the release criteria have been satisfied.

During the inspection, Survey Package A0345 was reviewed. A spot check of the release record for Survey Package A0345 indicated that the final, processed values presented for the surface activity measurements were technically accurate. Inspectors noted that the text material for the release records had to be generated, manually, by the computer operator. A future computer program upgrade may eliminate the need to manually generate the text material.

In summary, the licensee's method of collecting and analyzing the data appeared to be state-of-the-art. In addition, the licensee was working with the survey instrument manufacturer to continuously upgrade the data collection and processing techniques.

2.3.2 Quality Control of the Data Reduction Activities and Procedures

Section 3.5.1.1 of the Final Survey Plan, "Software Control," stated that the methods for internal-use software validation, verification, and control would be detailed in the FSV Decommissioning Project Radiation Protection Manual. Administrative procedure FSV-RP-ADM-I-105, "Control of Internally Developed Computer Programs," Revision 2, provided guidance on validation, verification, and control of the computer programs used for the final site survey.

The Procedure FSV-RP-ADM-I-105 was used to validate and verify the computer programs "Survey Package Database" and "DBACORR." The Survey Package Database was used to store information about the survey package. This program provided input used by DBACORR during the generation of the release records. Inspectors found that the licensee performed daily and weekly backups of the database. The database engineer maintained the computer programs and computer database.

Strengths and weaknesses were noted with the software control program. Up-front validation and verification testing of the programs were a strength, and requirements for future, in-process validation and verification were areas in need of potential improvement. For example, validation, verification testing, and modifications of computer programs and program modifications were performed in accordance with the procedure. However, the Procedure FSV-RP-ADM-I-105 did not specifically require routine testing of the computer programs. Inspectors determined that routine computer software testing could be an effective quality control method of ensuring the reliability of the data reduction activities and the accuracy of the data. The licensee stated that the instructions for performing routine software testing would be added to the applicable procedure in the near future.

Inspectors found several procedures that had not been updated. Procedural steps did not completely agree with the actual method of inputing data into the computer. For example, during the performance of Section 5.6 (Entering Survey Information) of Procedure FSV-RP-INST-I-234, "Download of the Ludlum Model 2350 and 2350-1 Data Loggers," Revision 5, the computer operator had to deviate from the procedure instructions to properly input the station number (location of computer terminal) into the spreadsheet.

A second procedure, Technical Basis Document FSV-FRS-TBD-206, "Background Determination," Revision 0 was not updated. Table 5-1 of this document listed the background values for selected materials (such as concrete, asphalt, tile) that were to be subtracted from the total surface activity measurements. Survey Package A0345 used a value for fiberboard ceiling tiles which was not listed in Table 5-1. These findings were pointed out to the licensee who

planned to update the procedures or the software, as appropriate, in the near future.

2.3.3 Summary

Overall, the licensee had developed and implemented a state-of-the-art program for data collection and analysis. The absence of a routine quality control program for validation of computer software was identified as an area of potential improvement. Several procedures had not been updated to reflect the latest methods for inputing data and selecting material background values.

2.4 Records Management of Survey Packages

2.4.1 Discussion: Survey Package Groupings and Tracking

In order to perform the final surveys, the plant was subdivided into discrete sections. With the exception of the Repower Area which was released in May 1995, the final survey packages were grouped as follows:

GROUPS	NO. OF PACKAGES
A: Buildings outside the Restricted Area	12
B: Buildings inside the Restricted Area	18
C: Turbine building and building No. 10	62
D: Unaffected Plant Systems	23
E: Effluent Discharge Flow Path	9
F: Reactor building	154
G: Affected Plant Systems	14
H: Embedded pipe & Inside the Reactor Vessel	9
I: Open land and miscellaneous building and systems	16

The licensee used the following two tracking system documents to maintain the status of each the 317 survey packages:

- "Final Breakdown Structure," Revision 1
- "Survey Package Tracking System"

The Final Breakdown Structure updated each survey grouping, unit, package number, and classification (Affected or Unaffected), and included a map of each survey area. The Survey Tracking System provided an up-to-date status

summary of each survey group, unit, and package. The four parts of the group Survey Tracking System included the following:

- Part I: Percent of Survey Preparation Complete
- Part II: Percent of Package Initiation Complete
- Part III: Percent of Package Implementation Complete
- Part IV: Percent of Data Analysis and Package Closure Complete

As of January 22, 1996, Group A packages were 87 percent complete; Parts I, II, and III were 100 percent complete; and Part IV was 64 percent complete. The Group A summary remarks explained that data analysis was in progress for 8 out of the 12 packages. Four Group A packages were waiting for final closure. These package were not finalized at the time of the inspection because the licensee had not resolved the technical issue of exposure rate background. Without a final determination on the exposure rate issue, the release records could not be completed for the exposure rate measurements. Only Groups B and C had an overall completion of Parts I-IV in excess of 50 percent.

2.4.2 Final Survey Package Review

The following Implementing Procedures were used to develop final survey packages and data:

- FSV-SC-FRS-I-102: Survey Design and Package Preparation

- FSV-SC-FRS-I-110: Final Survey Data Analysis

- FSV-SC-FRS-I-114: Survey Design and Package Preparation for Plant Systems

A "survey package" was prepared for each survey area. Survey packages included information in a standardized format to control and document radiation measurements taken during the final survey. Survey packages included survey instructions, a grid map, measurement data sheets, and a history of the area. History files were a compilation of information prepared for use in planning the survey, which summarized the operational history, characterization survey data, operational surveys, and other information as appropriate. History files were used to support the classification of each survey unit as either affected or unaffected.

Inspectors examined the following four final survey packages:

- A0341: Simulator Room - Suspect Affected
- A0342: Classroom - Suspect Affected
- A0344: Simulator Room - Non Suspect Affected
- A0345: Classroom - Non Suspect Affected

These four final survey packages only needed resolution of exposure rate issues to be 100 percent complete. The content of the four packages reviewed

met the requirements of licensee procedures and the Final Survey Plan. The inspectors noted one area for improvement with respect to the survey packages. Specifically, the details of the packages were difficult to follow as stand-alone documents. For example, a table of contents at the beginning of each final survey package would be an enhancement. Nonetheless, inspectors determined that the final survey packages were adequate.

2.4.3 Final Survey Package Investigations

The licensee's latest Final Survey Plan contained the criteria that was approved by the NRC on June 28, 1995, for reclassification and investigation of final radiological survey areas. Implementing Procedure FSV-SC-FRS-I-110, "Final Survey Data Analysis," Section 5.12 contained survey package investigation and reclassification criteria. Instructions for preparing replacement survey packages and documenting investigations were found in Implementing Procedure FSV-SC-FRS-I-102 "Survey Design and Package Preparation," Section 5.11. Section 5.11 of Procedure FSV-SC-FRS-I-102 requires, in part, that survey packages with reclassification investigations have a code "G" designation.

Inspectors reviewed the following eight survey packages to determine if the packages were properly investigated and reclassified:

- A0346: Buildings 107, 110, 1001, 1002, and 1015
- A0349: Buildings 7, 106, 111, 1000, and 1020
- B005: Buildings 13, 14, and 1014
- B006: Buildings 15, 16, 17, and 1003
- C009: Turbine Lube Oil Storage Tank Room
- C010: Service Water Booster Pumps Area
- C021: Level 6 Northwest General Area
- C052: Condensate Storage Tanks Mezzanine

Inspectors examined in greater detail survey packages, A0346, A0349, B006, C021, and C009 to evaluate compliance with the administrative action levels listed in FSV-SC-FRS-I-110, "Final Survey Data Analysis." Final survey results above these action levels are required to be investigated to determine whether additional sampling, survey unit reclassification, or remediation are required. In addition, the extent of the investigations were evaluated to determine if the investigations were ALARA and represented sound engineering judgement. For the five survey packages reviewed, all of the results above the action levels prescribed in FSV-SC-FRS-I-110 were investigated by PSCo. For Package Survey A0346, an investigation was conducted even though the particular survey result was below the action level. The investigations were, generally, well documented in the packages and provided additional confidence that the survey program was identifying significant contamination.

Two minor administrative/documentation problems were identified during the review of the above survey packages. In Package Survey A0346, the release record, and download file No. 730 indicated that the maximum beta

contamination level was 2459 dpm/100 cm^2. However, the investigation documentation indicated that the maximum contamination level was 2383 dpm/100 cm^2. A second documentation problem was identified in Survey Package C009. Attachment 6.3 in Survey Package C009 was reviewed by the Lead Engineer before the technical projects supervisor. This appeared to be contrary to FSV-SC-FRS-I-102, which indicates that it is the lead engineers responsibility to provide final approval, after the technical projects supervisor.

2.4.4 Investigations and Hard to Detect Nuclides

Survey Packages A0346, B006, and C009, described unaffected areas where investigations were performed. The Final Survey Plan stated that survey results in unaffected areas will not be corrected for the contribution of hard to detect nuclides (HTDNs), such as Fe-55 and H-3. This was technically adequate since contamination from licensed material was not expected to be found in unaffected areas. However, contamination levels between 25 percent and 50 percent of the surface contamination guideline limits were reported in the three above survey packages. These results were statistically above background and appeared attributable to licensed radioactivity. Inspectors found that the licensee investigated the results greater than 25 percent of the guideline in accordance with the Final Survey Plan. However, the survey results were not corrected for HTDNs.

Inspectors concluded that the licensee should evaluate whether or not corrections for HTDNs are technically merited when final survey results in unaffected areas exceed 25 percent and are attributable to licensed material. Particularly, if a HTDNs corrected result exceeded 50 percent of the guideline, but the uncorrected result does not.

2.4.5 Deleting Invalidated Investigation Measurements from the Database

Three investigations, for Survey Packages B006, C021, and C009, concluded that the survey results under investigation were not validated by the investigation. The licensee also decided to remove the investigated measurements from the final survey database and not include the data in the final survey report. Licensee investigations included a review of the following:

- The contamination potential of the area where the measurement was performed.
- The actual measurement results that ensured the instrumentation was properly functioning and that the data was properly handled.
- The collection of additional measurements in the area of concern.

Based on the inspector's review of investigation documentation in the final survey packages and discussions with the licensee, it was unclear that adequate measurements were made at the same location where the initial survey

identified contamination. It was, also, unclear if the initial measurement could be reproduced.

The inspectors determined that the licensee's investigations, as documented, lacked an adequate justification to support invalidating the original survey result under investigation. Inspectors determined this area as weak unless investigation findings conclusively invalidated the original results. Inspectors recommended that the licensee evaluate whether investigations of suspect measurements were adequately conducted to justify removing the original measurement from the survey database.

2.4.6 Investigation and Reclassification of the B005 Survey Package

As a followup to an inspection performed September 25-28, 1995, (Report 50-267/95-04), Survey Packages B005 and B017 were reviewed by the NRC. The finding in the last report indicated that the classification of unaffected area Survey Unit B005 appeared to not be fully justified by the past operational history in the area such as the presence of a hot-soil lab and previously identified contamination in parts of Survey Unit B005. In addition, the NRC found that the extent of the survey unit area that was reclassified after contamination was identified as too limited. Subsequent to the NRC inspection, the entire electrical warehouse building was reclassified as affected by the licensee. The additional surveys performed as a result of the reclassification identified contamination in a localized area at 6777 dpm/100 cm^2. This result emphasizes the importance of using the ALARA principles and reasonable conservatism when classifying or reclassifying survey units.

2.4.7 Scan Survey Coverage

During the review of survey packages, the NRC identified an inconsistency between the scan survey coverage required in nonsuspect affected areas, and the scan survey coverage required in suspect-affected and unaffected areas. Procedure FSV-SC-FRS-I-102 required scan surveys in the 1 meter squared (m^2) area around the direct measurement locations in a nonsuspect affected area. Since the survey coverage in nonsuspect affected areas was a minimum of one measurement per 20 m^2, the scan survey coverage in nonsuspect affected area translated to 5 percent of the survey unit surface area. Five percent scan coverage was inconsistent with the required scan survey coverage of 25 percent in unaffected areas and 100 percent in suspect-affected areas. Inspectors found that the licensee's scan coverage in nonsuspect affected areas was in compliance with the statements in the Final Survey Plan. However, Implementing Procedure FSV-SC-FRS-I-110 appeared to minimize the extent of scan survey coverage and may not represent appropriate ALARA practices.

Inspectors questioned whether the scan survey coverage percentage in nonsuspect affected areas should be at least that required for unaffected areas, and depending on the contamination potential in a given survey unit, possibly greater. Inspectors concluded that the licensee should perform an ALARA engineering evaluation to support their position on this matter.

2.4.8 Conclusions

Generally, the survey program and survey packages were documented adequately. Survey package investigations and reclassification were adequate. However, several technical concerns were identified during reviews of final survey packages whereby the licensee made decisions without clearly providing a documented ALARA basis. Inspectors identified the following concerns: (1) whether corrections for hard to detect nuclides should be made to survey results in unaffected areas that exceed 25 percent of the surface contamination limits; (2) whether investigations of suspect measurements were adequately conducted to justify removing the original measurement from the survey database; and (3) whether the scan survey coverage percentage in nonsuspect affected areas should be increased. Previous NRC Inspection Report 50-267/95-04 raised general concerns about the licensee's application of ALARA principles and conservatism, and this current report identified specific ALARA concerns. The inspectors recommended that the licensee address the specific concerns raised in this section of the report.

2.5 Quality Assurance/Quality Control

Quality assurance/quality control activities were reviewed to ensure that the commitments made in the Final Survey Plan were incorporated into the final survey program.

Section 3.5 of the Final Survey Plan, "Quality Assurance" required the licensee to have a Decommissioning Project Quality Plan that is based on 10 CFR Part 50, Appendix B. Section 3.5.1(i) of the Final Survey Plan required the licensee to have a third independent party to perform confirmation surveys on final survey packages, and required that survey packages be independently reviewed.

Section 3.5.2 of the Final Survey Plan, "Final Survey Quality Control Procedure," stated that a final survey procedure contained in the decommissioning project radiation protection manual will establish the quality activities not addressed in other procedures. The Final Survey Plan stated that these activities may include the conduct of quality control verification sampling and survey measurements, verification of survey measurement data, testing of computer calculations, documentation of surveys, and custody of instruments, samples and measurement data.

The licensee's administrative and implementing procedures were reviewed to ensure that the requirements established in the plan had been incorporated into the plant procedures. Inspectors found that Implementing Procedure FSV-SC-FRS-I-109, "Final Survey Quality Control Measurements," Revision 1, provided guidance for the performance of the quality control verification and sampling survey measurement. Procedure FSV-SC-FRS-109 required the licensee to resurvey 5 percent of the final survey packages.

Inspectors met with the licensee and their contractor Scientific Ecology Group to discuss the quality assurance program implementation within the Final

Survey Plan. Scientific Ecology Group implements the licensee's Final Survey Plan. The licensee stated that the Scientific Ecology Group will conduct the resurvey, and those survey packages will be readily identified. Inspectors reviewed the licensee's process for performing the 5 percent quality control check of final survey packages. Additionally, the licensee stated that General Public Utilities Nuclear Corporation would start implementing the independent third party verification program in February 1996.

Generally, the licensee had developed and implemented quality control procedures that met the intent of Section 3.5.2 of the Final Survey Plan. Other administrative and implementing procedures were used to incorporate the remaining requirements of Section 3.5.2, including procedures for sampling chain of custody, control of survey instruments, package preparation, and data review checklists. Inspectors stated that a more extensive inspection of the licensee's quality assurance and quality control program for the Final Survey Program would be conducted in the near future.

2.6. Plant Tours

During the plant tours, a technician was noted to be performing survey work on the turbine operating deck floor (Level 7). This technician was questioned about the work in process. This technician demonstrated knowledge and understanding of the task being performed but could not recall the details of the implementing procedure that he was working under. This finding suggested that some individuals apparently do not have detailed knowledge of the implementing procedures, but do understand the basic concepts established by the procedures.

According to Section 3.8.14 of the Final Survey Plan, access control measures were required to be implemented to protect areas from contamination subsequent to the final survey. During an inspection conducted in December 1995 (documented in NRC Inspection Report 50-267/95-05), tours of the plant were performed, in part, to monitor for the effectiveness of the implementation of the final survey isolation and control measures. In the previous inspection, the NRC concluded that the licensee's use of isolation and control boundaries to control areas that had been surveyed was inconsistent with the Final Survey Plan and the plan's implementing procedures.

During this inspection, plant tours were again performed, in part, to assess the effectiveness of the licensee's isolation and control boundary markers. In general, the access control measures had been upgraded in the reactor building. New and improved boundary markers were installed to control access to selected areas of the reactor building. Also, problem areas identified during the previous NRC inspection had been corrected by the licensee.

However, inspectors observed what seemed to be an inappropriate boundary marker in the turbine lube oil room of the turbine building. Licensee personnel stated that this modified final survey boundary marker was installed to control access to the room's ventilation system. Inspectors determined that the boundary marker was appropriate to control access to a plant system.

In summary, the effectiveness of the final survey access control measures was adequate. Improvement was noted in the reactor building with respect to boundary postings.

3 FOLLOWUP (92701)

3.1 (Closed) Inspection Followup Item 50-267/9504-01: Review of the Final Survey Plan implementation.

The September 1995 inspection plan was not fully implemented, because the licensee's Final Survey Plan was not fully implemented. Inspection items not fully reviewed during the previous team inspection included:

- Analysis of survey results and data reduction activities
- Review of completed survey packages to ensure compliance with the program requirements and that each survey package was a stand-alone auditable document.
- Implementation of the commitments made in the June 28, 1995, letter to the NRC with respect to the classification and investigation criteria.
- Review of the instrument sensitivities.
- Implementation of the final survey quality control activities, and development of the final survey quality control procedure.
- Review of records management, including the control of computer databases of the final survey data, the quality control oversight of these databases.

During this inspection, inspectors found that each of the above listed items had been sufficiently implemented by the licensee. Each item was inspected and is addressed in this inspection report.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

*T. Borst, Radiation Protection Manager
*S. Chesnutt, Senior Project Assurance Engineer
*M. Holmes, Project Assurance Manager
D. Seymour, Senior Quality Assurance Engineer

1.2 Contractor Personnel

*R. Argall, Radiochemistry/Training Supervisor Scientific Ecology Group (SEG)
D. Blain, Field Operations Coordinator, SEG
*M. Buring, Radiation Protection Operations Supervisor, SEG
*C. Cummin, Radwaste Supervisor
B. Dyck, Licensing Engineer, Westinghouse
*T. Howard, Project Director, Westinghouse
*W. Hug, Operations Manager, MK-Ferguson
*B. Mann, PSCo Project Assurance Consultant
*M. Miles, Field Operations Coordinator
R. McGinley, ALARA Supervisor, SEG
D. Parsons, Radiological Engineer, SEG
*G. Policastro, Technical Support Projects Supervisor, SEG
*J. Rood, Final Survey Lead Engineer, SEG
H. Story, Project Radiation Protection Manager, SEG
*M. Zachary, Final Survey Operations Supervisor, SEG

1.3 State of Colorado

*K. Weaver, Department of Public Health & Environment

1.4 Oak Ridge Institute for Science and Education

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

1.4 NRC, Office of Nuclear Materials Safety and Safeguards

*D. Fauver, Senior Project Manager, Division of Waste Management
D. Moser, Health Physicist, Division of Waste Management
*C. Pittiglio, Senior Project Manager, Division of Waste Management
M. Weber, Branch Chief, Division of Waste Management

1.5 NRC Region IV Personnel

*L. Carson II, Health Physicist, Division of Nuclear Materials Safety
*R. Evans, Health Physicist, Division of Nuclear Materials Safety
R. Scarano, Director, Division of Nuclear Materials Safety
*B. Spitzberg, Branch Chief, Division of Nuclear Materials Safety

* The personnel listed above attended the exit meeting. In addition to the personnel listed above, the inspectors contacted other members of the site staff during this inspection.

2 EXIT MEETING

An exit meeting was conducted on January 25, 1996. During the meeting, the inspectors reviewed the scope and findings of the inspection. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

Inspectors identified an Unresolved Item based on ORISE's preliminary side-by-side comparisons which identified variations in the licensee's shielded measurements results. The observed variations could affect the validity of final survey data. Licensee representatives did not have a full explanation to account for the variations and how they would affect final survey results. The licensee was requested to submit the results of their side-by-side and scan survey measurements, performed simultaneously with the ORISE representatives and an explanation of the shielded measurement variations. The licensee requested the NRC to submit ORISE's instrument correlation data. NRC personnel indicated that they would provide the information.

Inspectors expressed concern that resolving issues associated with the embedded piping measurement methods and the background exposure methods could restrain the NRC's ability to review the Final Survey Reports in a timely manner. NRC personnel indicated that they would provide formal comments on the licensee's embedded piping submittal dated January 18, 1996, by the end of this month. Licensee management stated that they would provide the exposure rate background analysis, Technical Basis Document, by the end of February 1996.