ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-267/95-05

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: December 18-20, 1995

Inspectors: R. J. Evans, Health Physicist
Nuclear Materials Inspection & Fuel Cycle/Decommissioning Branch

L. C. Carson II, Health Physicist
Nuclear Materials Licensing Branch

Approved: D. Blair Spitzberg — 1-23-96

D. Blair Spitzberg, Ph.D., Chief
Nuclear Materials Licensing Branch — Date

Inspection Summary

Areas Inspected: Routine, announced inspection of plant status, operational safety verification, radiological environmental monitoring program, occupational exposure control during decommissioning activities, and followup of previously identified inspection findings.

Results:

- Since the previous inspection, housekeeping in the reactor building had improved, but had declined in the turbine building. Housekeeping in the Independent Spent Fuel Storage Installation was adequate. Radiological postings met the requirements of 10 CFR Part 20 (Section 2.1).

- The use of final survey isolation and control boundaries to control areas that had been characterization surveyed was noted to be inconsistent with the licensee's final survey plan and the plan's final survey implementing procedures. No safety concern was identified with this observation (Section 2.1).

- The licensee exited a Decommissioning Technical Specification condition when the shield water system was drained from the prestressed concrete reactor vessel. This was the third Decommissioning Technical Specification general requirement that the licensee has considered to be no longer applicable at the site (Section 2.2).

- The inspectors reviewed the licensee's investigation into an employee's claim of having been contaminated while at the facility. The inspectors agreed with the licensee's conclusion that the employee had not been contaminated as initially suspected by the employee (Section 2.3).

- The licensee had implemented a radiological environmental monitoring program that met the Decommissioning Technical Specifications. The 1994 Annual Radiological Environmental Operating Report was an improvement over the 1993 Report (Section 3.1).

- The 1994 Annual Radioactive Effluent Release Report for 1994 met the Decommissioning Technical Specifications (Section 3.2).

- Based on a review of the Radiological Occurrence Report Program concerning turbine building contamination, the inspectors concluded that weaknesses existed in fully documenting corrective actions and providing complete assessments of radiological issues (Section 4.1).

Summary of Inspection Findings:

- Deviation 267/9302-01 was closed (Section 5.1).
- Violation 267/9405-01 was closed (Section 5.2).
- Violation 267/9501-01 was closed (Section 5.3).

Attachment:

- Persons Contacted and Exit Meeting

DETAILS

1 PLANT STATUS

The major tasks in progress at the facility were the dismantling and decontamination of the radioactive portions of the prestressed concrete reactor vessel, decontamination of secondary systems, and work related to the final survey of the site. The decommissioning work was being performed by a Westinghouse Team that consisted of personnel from Westinghouse Electric, Scientific Ecology Group, and MK-Ferguson.

Since the last plant status update, documented in NRC Inspection Report 50-267/95-01, the activities completed by the licensee include draining and disposal of the prestressed concrete reactor vessel shield water, removal of the temporary shield water system components, removal of the steam generators and helium circulators, and removal of 28 blocks of activated concrete (beltline concrete cuts) from the prestressed concrete reactor vessel cavity.

Work in progress at the time of the inspection included the dismantling and decontamination of the liquid radwaste system; building drains; and heating, ventilation, and air conditioning subsystems. Other work in progress included the removal of the lower plenum insulation, decontamination of the bottom head/lower plenum area of the prestressed concrete reactor vessel, decanting and barrelling of concrete cutting slurry, and the final survey.

One position change occurred recently in the licensee's staff. The radiation protection manager assumed the dual positions of decommissioning program director/radiation protection manager on November 20, 1995. The former decommissioning program director was reassigned to the position of team leader for a corporate office special project, reporting directly to the vice president of engineering and operations support.

2 OPERATIONAL SAFETY VERIFICATION (71707)

2.1 Plant Tours

Routine tours 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 inspected included the maintenance of the radiologically restricted areas, housekeeping, and material control. Also inspected was the control of radiological postings, labelling, and boundaries.

During this inspection, housekeeping was noted to be improving in the reactor building but declining in the turbine building. This observation was the reverse of previous plant tours (refer to NRC Inspection Report 50-267/95-01). The removal of loose material from the reactor building, in part, has resulted in the overall improvement of housekeeping in the reactor building. Plant equipment removal activities in the turbine building, in part, have resulted in an overall decline of housekeeping in that building. In addition, the

Independent Spent Fuel Storage Installation was toured and housekeeping was noted to be adequate at that facility.

Licensee compliance with the Decommissioning Technical Specifications was also inspected during the plant tours. No examples of noncompliances were identified. One minor procedure problem was noted and reported to the licensee. The equipment operator round sheet, used by operators during routine building tours, referenced several out-of-date sections of both the Decommissioning Technical Specifications and the Offsite Dose Calculation Manual.

Radiological postings and boundaries were noted to be in compliance with 10 CFR Part 20. In addition, the licensee's isolation and control measures, used to isolate areas of the plant that had been final surveyed, were inspected.

According to the NRC-approved final survey plan, measures will be implemented by the licensee, or its contractors, to protect areas from being recontaminated subsequent to the performance of the final survey. These measures include the use of barriers to control ingress and egress and the installation of postings requiring personnel to perform contamination monitoring prior to entering these areas. The isolation and control measures were proceduralized in the Implementing Procedure FSV-SC-FRS-I-108, "Final Survey Isolation and Control Measures," Revision 3.

The isolation and control measures were adequate outside of the reactor building. However, inside of the reactor building the isolation and control measures were not being strictly adhered to. In addition, the isolation and control measures were being used differently inside of the reactor building than outside of the reactor building.

During the tour of the reactor building, the green isolation and control boundaries were being used to isolate areas that had been characterization surveyed (but not final surveyed) from the remainder of the areas in the reactor building. The licensee was attempting to control the movement of radioactive material into these areas by using the final survey isolation and control measures.

Several areas that were "isolated" were noted to not be consistent with the isolation and control measures that had been established by the licensee in the implementing procedure as follows:

- Radioactive material was identified in one isolated area. A vacuum cleaner labelled as "radioactive material" and "internal contamination" was inside of a controlled area on Level 6 of the reactor building.

- Boundaries were not in place (for example, the isolation rope was on the ground and not in place across ingress/egress routes), were not obvious (no isolation rope present), or were nonexistent in at least ten places in the reactor building.

- Hoses and cables traversed through the boundaries at several locations; the condition of the hoses (surveyed or not surveyed) was not obvious.

- Isolation boundary labels that were controlled by procedure were noted to have been marked up in the plant with instructions that were not authorized by procedure.

In addition, the procedure requirements that all personnel were required to perform a whole-body frisk and that equipment had to be verified free of radioactive contamination prior to entering an isolated area were not being adhered to.

In conclusion, the use of final survey isolation and control boundaries in the reactor building to control areas that had been characterization surveyed was not consistent with the isolation and control measures that had been established by the licensee in the implementing procedure.

Despite the inconsistent application of the isolation and control boundaries, a safety concern did not exist, because the entire reactor building was still a radiologically restricted area. All equipment and personnel were required to be verified free of contamination prior to exiting the building. The misuse of boundaries within the reactor building did not circumvent the requirement that all people and items were to be verified free of radioactive contamination prior to exiting the reactor building.

2.2 Exiting of a Decommissioning Technical Specification Requirement

Shield water was added to the prestressed concrete reactor vessel to provide radiation shielding and contamination control during decommissioning activities. The volume in the prestressed concrete reactor vessel previously has been about 350,000 gallons, although the amount was reduced as components were removed from the reactor cavity. The licensee recently drained the remaining shield water from the prestressed concrete reactor vessel and exited the Decommissioning Technical Specifications requirement related to the shield water volume.

Decommissioning Technical Specifications LC 3.4, "Prestressed Concrete Reactor Vessel Shielding Water Tritium Concentration," states, in part, that the tritium concentration in the shield water shall not exceed 62.4 microcuries per cubic centimeter whenever there was shield water within the prestressed concrete reactor vessel. The value of 62.4 microcuries per cubic centimeter correlated to a maximum tritium source term of 100,000 curies. This maximum amount of tritium would ensure adequate protection to members of the public following an accidental release of the water volume in the reactor building.

On October 16, 1995, the final volume of shield water was removed from the prestressed concrete reactor vessel. Although small amounts of water may remain in the bottom of the prestressed concrete reactor vessel from decommissioning and decontamination activities, the amount of tritium in the remaining water volume was not expected to approach the maximum concentration that was assumed in the accident analysis considerations.

The licensee concluded that as of October 16, 1995, the license condition for tritium concentration in prestressed concrete reactor vessel shield water was no longer applicable and all surveillance testing activities related to Decommissioning Technical Specification LC 3.4 were discontinued. The licensee previously exited two other Decommissioning Technical Specification requirements, reactor building confinement integrity (LC 3.1), and reactor building ventilation exhaust system (LC 3.2). Both were exited on March 21, 1994, when the final shipment of activated graphite components was removed from the site (refer to NRC Inspection Report 50-267/94-03). The licensee discussed these actions with the NRC prior to implementation. The NRC inspectors had no concerns with the licensee's decision.

A review of several implementing procedures revealed that all references to the shield water system have not been eliminated from plant procedures, although the water has been drained and the support equipment has been removed.

2.3 Potential Contamination Incident

On November 9, 1995, at 10:45 p.m., the control room was notified that an individual had walked into the St. Anthony's North Hospital, located in Westminster, Colorado, claiming to have radiation burns and potentially contaminated clothing. According to documents retained by the licensee, the individual, a new MK-Ferguson employee, walked into the hospital at about 10:30 p.m. The individual thought he had radiation burns on his left forearm and possessed potentially contaminated clothing. In response to the occurrence, the NRC Operations Center was notified of the incident within 4 hours in accordance with 10 CFR 50.72.

While at the hospital, a skin condition similar to a sunburn (reddening of the skin) was noted on the individual, although no blistering of the skin was observed. The individual also claimed that the skin irritation/pain was more pronounced when his coat was on versus when the coat was off.

The patient was isolated, and the Westminster Fire Hazmat team was called by the hospital. The team performed a radiological survey of the person and did not identify any radioactive contamination on the clothing or the individual. However, the team was not confident of the instrument calibration or sensitivity; therefore, a response team from the Rocky Flats facility was called to the hospital. The second team performed a radiological survey of the individual. The individual and his coat were both found to be free of radioactive contamination.

A security and health physics record review was performed by the licensee to determine the in-plant activity of the individual. Records revealed that the individual was at the site on Monday, November 6, 1995, for orientation training. The individual did not enter the plant that day. The next day, the individual was inside of the plant perimeter fence, but outside of the radiologically restricted area for 49 minutes. The individual had received a whole-body count during this time frame.

During the whole-body count, the individual apparently laid his work coat on a tray near some "vials." Following the whole-body count, the individual left for the day and did not come in for work the following day (November 8, 1995).

The licensee took several corrective actions immediately following the discovery of the potentially contaminated individual. The licensee performed a radiological survey of the whole-body count room on November 10, 1995. No loose radioactive contamination was identified. The portal monitors, located at the exit of the security-restricted area, were verified to be operable and in service (the individual had to exit the plant through the portal monitors; the monitors should have activated a local alarm if the monitors detected that the individual was contaminated). In addition, the individual's whole-body count results were reviewed. The results were negative, indicating that no abnormal radioactive materials were identified during the count. The vials that the individual mentioned were subsequently determined to be containers of "Ultima Gold," a chemical compound used in liquid scintillation counting equipment. The compound did not contain radioactive or hazardous ingredients, although the compound does contain a skin irritating solvent.

On November 11, 1995, the licensee contacted the individual. He claimed that he was doing fine, the "problem" had gone away and had not returned, and that he had thrown the coat away. The individual rejected the licensee's offer of a repeat whole-body count.

In summary, the cause of the skin irritation was not clearly identified by the licensee. Following a review of the employee's locations on site prior to the development of his condition, and the results of radiation surveys and monitoring of the individual, the inspectors agreed with the licensee's conclusion that the employee's skin condition was not caused by radioactive contamination that had originated from the Fort St. Vrain site.

2.4 Conclusions

Since the previous inspection, housekeeping in the reactor building had improved, but had declined in the turbine building. Housekeeping in the Independent Spent Fuel Storage Installation was adequate. Radiological postings met the requirements of 10 CFR Part 20.

The use of final survey isolation and control boundaries to control areas that had been characterization surveyed was noted to be inconsistent with the licensee's final survey plan and the plan's final survey implementing procedures. No safety concern was identified with this observation.

During October 1995, the licensee exited a Decommissioning Technical Specification condition when the shield water system was drained from the prestressed concrete reactor vessel. This was the third Decommissioning

Technical specification general requirement that the licensee has considered to be no longer applicable.

The inspectors reviewed the licensee's investigation into an employee's claim of having been contaminated while at the facility. The inspectors agreed with the licensee's conclusion that the employee had not been contaminated as initially suspected by the employee.

3 RADIOLOGICAL ENVIRONMENTAL MONITORING (80721)

An inspection of the radiological environmental monitoring program was performed to ensure that the program was being effectively implemented by the licensee.

3.1 Review of Annual Radiological Environmental Operating Report

Environmental monitoring consisted of the collection and analysis of samples of air, water, soil, foodstuff, biota (animal and plant life), and other media from the area around the site. Environmental monitoring has been performed, in part, to demonstrate facility compliance with applicable standards and to assess the effects, if any, of the facility on the local environment.

The licensee was required to have a radiological environmental program to comply with the requirements of the Decommissioning Technical Specifications, Section 5.4.4.b. Attachment F to the Offsite Dose Calculation Manual, a supporting document of the Decommissioning Technical Specifications, provides the description of the radiological environmental monitoring program requirements.

A detailed review of Attachment F and the 1994 Annual Radiological Environmental Operating Report was performed. A few minor discrepancies were noted between the manual and the report. For example, the Offsite Dose Calculation Manual allows deviations from the manual requirements as long as the changes are documented in the Annual Radiological Environmental Operating Report. The 1994 Report did not clearly explain inconsistencies related to the locations of the seven airborne samples. The inconsistencies included a 45 versus 30 kilometer limit for control locations and agreement on which location had the highest historic calculated annual average airborne "X/Q." Also, the paragraphs on pages 49 and 51 of the Offsite Dose Calculation Manual were worded in a confusing manner.

Aside from the minor discrepancies specified above, the 1994 report was noted to be an improvement over the 1993 report. The number and types of discrepancies were fewer and less significant in the 1994 report than in the 1993 report.

3.2 Review of Annual Radioactive Effluent Release Report

The 1994 Annual Radioactive Effluent Release Report was reviewed. The report summarized the gaseous effluent, liquid effluent, and solid waste released from the site during calen year 1994. The licensee concluded that the releases were within the li..ts of 10 CFR Parts 20 and 50 as well as 40 CFR 190. No specific concerns were identified during the review of the effluent release report.

Surveillance Procedure SR-ODC-6.3-72, "Reactor Building Sump Effluent Release," Revision 8, was reviewed. The purpose of this procedure is to ensure that the liquid releases from the reactor building sump were consistent with the Offsite Dose Calculation Manual. Two minor discrepancies were identified in this procedure. First, the procedure was worded such that a release can begin with inoperable equipment. The Offsite Dose Calculation Manual states, in part, that releases can continue (not begin) with inoperable equipment. A licensee representative stated that releases had not been started with inoperable equipment.

Second, the Offsite Dose Calculation Manual, Step 6.2.6, stated, in part, that a channel check of flow instrumentation consisted of verifying indication of flow during periods of release. This step was considered ambiguous, because neither the inspector nor a control room operations person clearly understood how this step was effectively implemented. Additional manual guidance may be appropriate to ensure that this step is performed correctly during future liquid releases.

3.3 On-site Meteorological Monitoring Program

During an inspection conducted in July 1994 (documented in NRC Inspection Report 50-267/94-05), the site's meteorological monitoring equipment was examined. At that time, the licensee was noted to be properly operating and maintaining the equipment.

On April 17, 1995, the licensee submitted proposed changes to the emergency response plan to the NRC for review and approval. The proposed changes included a deletion of the requirement to obtain and use meteorological information in dose assessments. The NRC concluded that the proposed changes to the emergency plan did not degrade the effectiveness of the current emergency response capability of the licensee. Therefore, the NRC approved the licensee's changes to the Emergency Response Plan on November 27, 1995, which eliminated the licensee's requirement to maintain the meteorological monitoring equipment in an operable condition.

3.4 Conclusions

The licensee had implemented a radiological environmental monitoring program that met the intent of the Decommissioning Technical Specifications. Although a few minor discrepancies were identified, the Annual Radiological Environmental Operating Report for 1994 was an improvement over the 1993 Report. Minor discrepancies were also identified with one surveillance procedure. Overall, the annual effluent report also met the intent of the Decommissioning Technical Specifications.

4 OCCUPATIONAL EXPOSURE DURING SAFSTOR AND DECON (83100)

4.1 Audit, Appraisal, and Self-Assessments

Inspectors reviewed the licensee's self assessment program for identifying, evaluating, and documenting significant radiological occurrences and assuring that timely corrective actions were implemented. This inspection consisted of selected interviews with plant personnel, independent verifications of plant activities, and reviews of Radiological Occurrence Reports. Inspectors reviewed Radiological Occurrence Reports that had been written from October though December 1995. Inspectors reviewed the licensee's contamination findings for consistency with the licensee's Procedure FSV-RP-SASP-I-101, Revision 2, "Fort St. Vrain Decommissioning Project Radiological Occurrence Reports."

Inspectors found seven Radiological Occurrence Reports that had been written in November 1995 concerning the discovery of radioactive contamination at various locations on Level 7 of the turbine building. Inspectors examined the following Radiological Occurrence Reports: 95-069, 95-070, 95-072, 95-074, 95-075, 95-077, and 95-078. Loose and fixed radioactive contamination levels measured from less than 1000 disintegrations per minute per 100 centimeters squared (dpm/100cm2) to 63,000 dpm/100cm2. Finding contamination in the turbine building was of concern to the inspectors for the following reasons:

- Turbine building Level 7 was originally classified as an "Unaffected Area" by the Decommissioning Plan initial contamination surveys and now would have to be reclassified as an "Affected Area."

- The radiation protection program's routine contamination surveys in 1995 had not detected any radioactivity on Level 7 of the turbine building where a "Clean Area" and lunch room existed.

According to the Radiological Occurrence Report Procedure, situations such as finding elevated levels of contamination did not necessarily warrant an Radiological Occurrence Report. However, the procedure recognized that minor contamination had to be documented to ensure that resurveys were conducted in accordance with the Decommissioning Final Survey Plan. The licensee classified the Radiological Occurrence Report contamination issues as minor deficiencies and consolidated several of the turbine building contamination Radiological Occurrence Reports into Radiological Occurrence Reports 95-069 and 95-070.

Inspectors determined that the licensee's Radiological Occurrence Reports appropriately identified concerns in the area of routine contamination surveys. Inspectors realized that the turbine building contamination was identified during decommissioning surveys. However, the Radiological Occurrence Reports did not document the corrective actions to be taken for updating the Decommissioning Final Survey Plan.

Radiological Occurrence Report 95-069 stated that areas would be reclassified because contamination was in excess of controlled area limits, and the areas would be remediated as part of the Decommissioning Final Survey Plan. Licensee management explained that specific procedures within the Decommissioning Final Survey Plan would specifically revaluate, reclassify, and document the turbine building contamination based on the Radiological Occurrence Reports. At the time of this inspection, final documentation regarding the reclassification of the turbine building level was incomplete.

During tours with licensee's representatives, inspectors observed several contamination markers on the turbine building floor on Level 7 and in the lunch area. Licensee's staff stated that the markers identified areas that would have to be decontaminated before the final decommissioning survey. Both Radiological Occurrence Reports 95-069 and 95-070 stated that contamination discovered in the turbine building did not represent a breakdown of the radiation protection program, and there were no radiological consequences. Inspectors determined that the licensee's Radiological Occurrence Reports were not complete, because the Radiological Occurrence Reports did not evaluate why the routine contamination survey program had not detected radioactivity on Level 7 of the turbine building. Within the Radiological Occurrence Reports, the licensee did not evaluate why contamination existed in the turbine building lunch area.

In response to the inspectors' concerns about the lunch area, the licensee stated the radioactivity was fixed contamination, visibly marked, and represented no radiological hazard. The licensee explained that the instrumentation and techniques used during decommissioning surveys were more thorough, and consequently, the areas could have gone undetected during routine contamination surveys. A potential existed that contamination could be transferred from the reactor building to the turbine building. So, as part of Radiological Occurrence Report 95-070, the licensee conducted a self-assessment on controlling access to Level 7 of the turbine building. The inspectors reviewed the self-assessment and found that no evidence supported contamination being transferred from the reactor building to the turbine building.

4.2 Conclusions

Based on a review of the Radiological Occurrence Report Program concerning turbine building contamination, inspectors concluded that weaknesses existed in fully documenting corrective actions and providing complete assessments of radiological issues. Licensee management stated that they would further evaluate the Radiological Occurrence Report Program's applicability during Decommissioning Final Survey Plan activities.

5 FOLLOWUP (92701)

5.1 (Closed) Deviation 267/9302-01: Inadequate Work Instructions

During an NRC inspection conducted in 1993, the inspectors concluded that the reactor building crane overload incident of May 27, 1993, was the result of inadequate work instructions. The failure to have adequate work procedures was determined to be a Deviation from commitments made to the NRC in the Decommissioning Plan.

Following the licensee's investigation of the incident, the licensee concluded that the root cause of the event was a breakdown in communications, not inadequate work instructions. The NRC subsequently agreed with the licensee but noted that the inadequate work instructions contributed to the event.

Corrective actions taken by the licensee included inspection of the crane, enhanced use of a load indicating device, revision of work instructions, and issuance of a memorandum to workers. Since the incident, the licensee has performed many heavy lifts with successful results, including the core support floor lift (documented in NRC Inspection Report 50-267/95-01) as well as the recent prestressed concrete reactor vessel concrete segment lifts. In addition, the work package for the recent concrete block lifts was reviewed and the corrective actions proposed in 1993 were included in the November 1995 work instructions.

5.2 (Closed) Violation 267/9405-01: Violation of Decommissioning Technical Specifications

During an inspection conducted in July 1994, a number of discrepancies were identified between the Offsite Dose Calculation Manual Program requirements and the 1993 radiological environmental monitoring program as implemented by the licensee's contractor. The failure to effectively implement all aspects of the program was identified as a violation of Decommissioning Technical Specifications.

During this inspection, the 1994 radiological environmental monitoring report was reviewed. Overall, the 1994 report was noted to be an improvement over the 1993 report with respect to Offsite Dose Calculation Manual compliance. Although a few minor discrepancies were noted in the 1994 report, the licensee was in compliance with the requirements established in the Offsite Dose Calculation Manual. The concerns identified with the 1993 report were corrected in the 1994 report.

5.3 (Closed) Violation 267/9501-01: Failure to Adhere to Procedure Requirements

During an inspection conducted in February 1995, a repeat violation of labelling requirements was identified. The licensee was inconsistently labelling radioactive material with sufficient information to alert personnel to the potential hazards of the material. One of the contributing causes of the repeat violation was the high number of items labelled as radioactive material in the reactor building.

During this inspection, tours were performed, in part, to determine if radioactive material was properly labelled in the plant.

Two minor discrepancies were identified and reported to the licensee for resolution. First, a small Department of Transportation container in the Turbine Building was labelled as radioactive material but was apparently empty. The container should have been labelled as "empty" or the radioactive material labels should have been erased or obliterated on the exterior of the container. Second, garbage bags were not labelled with sufficient information to alert personnel to the potential hazards of the material in the bags. This practice is understandable, because the radioactive hazards are usually not quantified until the bags are sealed and disposed of. However, garbage bags were not listed as one of the exceptions for labelling requirements in the Administrative Procedure FSV-RP-RAM-A-100, "Radioactive Material Control Program," Revision 4.

These two findings were considered isolated occurrences and were not representative of more widespread problems. Overall, the licensee was in compliance with the procedural requirements for labelling of radioactive material.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

T. Borst, Radiation Protection Manager
A. Crawford, Vice President
M. Fisher, Special Projects
M. Holmes, Project Assurance Manager
D. Seymour, Senior Quality Assurance Engineer

1.2 Contractor Personnel

R. Argall, Radiochemistry/Training Supervisor, Scientific Ecology Group
M. Buring, Radiation Protection Operations Supervisor, Scientific Ecology Group
D. Cummin, Radwaste Supervisor, Scientific Ecology Group
B. Dyck, Licensing Engineer, Westinghouse
W. Hug, Operations Manager, MK-Ferguson
V. Likar, Technical Services Manager, Westinghouse
R. McGinley, ALARA Supervisor, Scientific Ecology Group
M. Miles, Field Operations Coordinator, Scientific Ecology Group
G. Policastro, Technical Projects Supervisor, Scientific Ecology Group
G. Rood, Final Survey Lead Engineer, Scientific Ecology Group
M. Zachary, Final Survey Operations Supervisor, Scientific Ecology Group

1.3 NRC Region IV Personnel

L. Carson II, Health Physicist, Division of Nuclear Materials Safety
R. Evans, Health Physicist, 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 period.

2 EXIT MEETING

An exit meeting was conducted on December 20, 1995. 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.