U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-285/82-26 License No. DPR-40

Docket: 50-285

Licenses: Omaha Public Power District (OPPD)
1623 Harney Street
Omaha, Nebraska 68102

Facility Name: Fort Calhoun Station (FCS)

Inspection At: Fort Calhoun Station, Blair, Nebraska

Inspection Conducted: October 25-29, 1982

Inspectors: Ronald Baer, Radiation Specialist — Date 1/18/83

Harold Chaney, Radiation Specialist — Date 1/19/83

Wesley Holley, Radiation Specialist — Date 1/18/83

Approved By: for Blaine Murray, Chief, Facilities Radiation Protection Section — Date 1/18/83

Inspection Summary:

Inspection on October 25-29, 1982 (Report 50-285/82-26)

Areas Inspected: Routine, unannounced inspection of the licensee's radiation protection program during operations, transportation activities, and select NUREG-0737 items. The inspection involved 107 onsite inspector-hours by three NRC inspectors.

Results: Of the three areas inspected, no violations or deviations were identified. One unresolved item is discussed in paragraph 4.e. Several new open items are discussed in paragraph 3.

8303180244 830126
PDR ADOCK 05000285
G PDR

Details

1. Persons Contacted (OPPD)

*W. C. Jones, Division Manager, Production Operations
*R. L. Andrews, Section Manager, Operations
C. J. Brunnert, Quality Assurance Inspector
S. Crites, Senior Designer
D. Dale, Chief, Quality Assurance Inspector
D. Ecklund, Nuclear Design Engineer
J. Fluehr, Reactor Engineer
*G. K. Gambhiv, Manager, Generating Station Engineering (Nuclear)
*W. G. Gates, Plant Manager, FCS
*B. J. Hickle, Supervisor, Chemistry and Radiation Protection
B. Lisowyj, Supervisor, Quality Assurance
C. Mallory, Chemistry Radiation Protection Training Coordinator
J. Mattice, ALARA Coordinator
K. Miller, Test Engineer
*K. J. Morris, Manager, Administrative Services
M. Miroomand, Senior Design Engineer
*D. L. Patterson, Licensing Administrator, Production Operations
G. Peterson, Supervisor, Maintenance, FCS
A. W. Richard, Supervisor, Technical, FCS
*G. L. Roach, Plant Health Physicist
R. Short, Licensing Engineer
*P. M. Surber, Section Manager, Generating Station
F. Swihel, Engineering Training Instructor, Operations
C. Vanecek, Shift Supervisor, FCS
*M. C. Winter, Manager, Quality Assurance

*Denotes those present during the exit interview October 29, 1982.

The NRC resident inspector was not available for the exit interview due to a meeting at the Region IV office.

The NRC inspectors also interviewed radiation protection technicians, instrument and control technicians, and operations and administrative personnel.

2. Licensee Action on Previous Inspection Findings

(Open) Open Item (285/8016-33): Transportation - This item involved determination of the curie content of waste drums using contact dose rate measurements. The licensee had made no progress in resolving this item and additional misuse of the curie content conversion factor was identified. See paragraph 5.1(1)(d) of this report.

(Closed) Open Item (285/8004-03): Contaminated Material Control - This item involved a defective differential pressure (DP) gauge on the high efficiency filtered ventilation system of the solid waste compactor. The licensee had installed a working DP gauge on the ventilation system and instructions on DP limits. See paragraph 5.j of this report.

3. Open Items Identified During This Inspection

Open Item (285/8226-01): Radiation Worker Training - The licensee had not included all elements of Regulatory Guide 8.27 recommendations into this radiation worker training program. See paragraph 4.c for details.

Open Item (285/8226-02): Airborne Radioactivity Monitoring Program - The licensee did not use a collection media for airborne radioactivity measurements acceptable by industry standards to establish alpha concentrations. See paragraph 4.e for details.

Unresolved Item (285/8226-03): Radiation Protection Instrumentation Calibrations - The licensee was not able to provide calibration data on all radiation protection instrumentation calibrations to determine calibration frequency and calibration after maintenance. See paragraph 4.e for details.

Open Item (285/8226-04): Calibration of Whole Body Counter - The licensee had not prepared a procedure for the whole body counter consistent with ANSI-N343-1978. See paragraph 4.f for details.

Open Item (285/8226-05): Management Control of Transportation Activities - The licensee had not provided a suitable job description for personnel performing transportation activities; provided proper management review of day-to-day transportation activities; or implemented appropriate procedures that adequately cover all aspects of the licensee's transportation activities. See paragraph 5.a for details.

Open Item (285/8226-06): Selection of Radioactive Material Transporting Packages - The licensee had not implemented suitable procedures with appropriate quality assurance controls that will ensure that the proper selection and qualification of packages is performed for radioactive material shipments. See paragraph 5.b for details.

Open Item (285/8226-07): Preparation of Packages for Shipment of Radioactive Materials - The licensee had not implemented an adequate quality control program for the packaging of radioactive materials for shipment. See paragraph 5.c for details.

Open Item (285/8226-09): Delivery of Completed Packages to Carriers - The licensee had not implemented procedures that would ensure compliance with the waste shipment prior notification requirement of 10 CFR Part 71.5a. See paragraph 5.d for details.

Open Item (285/8226-08): Indoctrination and Training Program - The licensee had not developed an adequate program for training of solid waste process operators. See paragraph 5.h for details.

Open Item (285/8226-10): Licensee Audit Program - The licensee's audit team did not include a person knowledgeable in NRC and DOT transportation requirements on the audit team. See paragraph 5.i for details.

Open Item (285/8226-11): Examination of Licensee's Packages - The licensee had not implemented programs or reevaluate the effectiveness of existing programs for the reduction of generated radioactive solid waste and processing of stored wastes needing unique processes. See paragraph 5.j for details.

Open Item (285/8226-12): Radioactive Material Receipt and Shipment Records - The licensee's radioactive material shipping records did not provide for the verification that all NRC and DOT requirements have been complied with for radioactive material shipments. See paragraph 5.k for details.

Open Item (285/8226-13): Licensee Solid Waste Packaging and Transportation Activity Procedures - The licensee's procedures did not provide a concise and comprehensive program that will ensure compliance with all NRC and DOT regulations governing the licensee's activities involving transportation of radioactive material. See paragraph 5.1 for details.

Open Item (285/8226-14): NUREG-0737, Item II.B.3, "Postaccident Sampling Capability" - The licensee's Postaccident Sampling System (PASS) was not operable. See paragraph 6 for details.

Open Item (285/8226-15): NUREG-0737, Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 1, "Noble Gas Effluent Monitor" - The licensee's High Range Noble Gas Effluent Monitors were not installed and operable. See paragraph 6 for details.

Open Item (285/8226-16): NUREG-0737, Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 2, "Sampling and Analysis of Plant Effluents" - The licensee's iodine gaseous and particulate effluent monitors were not installed and operable. See paragraph 6 for details.

Open Item (285/8226-17): Calibration of Containment High-Range Radiation Monitors - The licensee's calibration procedures did not include calibration below 10 R/h with a calibrated radiation source. See paragraph 6.c(1)(c) for details.

Open Item (285/8226-18): Modifications for Control Room Habitability - The licensees had not completed modifications to satisfy NUREG-0737, Item III.D.3.4. See paragraph 6.e(3) for details.

4. Radiation Protection - Operations

   a. Radiation Protection Organization

   The Fort Calhoun radiation protection organizational structure is depicted by Figure 1. The chemistry and radiation protection (CARP) staff had an authorized level of 25 permanently assigned personnel, 23 were filled. The licensee rotated CARP technicians between chemistry and radiation protection assignments. The licensee had recently authorized a total of six CARP technician trainees, four of these positions had been filled at the time of this inspection with two remaining positions expected to be filled in the near future. The plant health physicist was assigned supervisory responsibility for the daily activities of the radiation protection section. He had assigned CARP technicians to perform radiation protection foremen duties, ALARA supervisor, and radioactive waste coordinator duties. Persons performing these duties were either senior technicians or technician-graded individuals who met the recommendations of ANSI-18.1-1971.

The licensee supplements the radiation protection section with five contractor health physics technicians. The NRC inspectors noted that the licensee had plans to obtain the services of about 25 contractor health physics technicians to assist with the 1983 refueling outage.

The NRC inspectors noted that the radiation protection section had not filled all authorized staffing positions and were concerned on the inordinate use of contractor health physics technicians to perform radiation protection activities during normal plant operation.

No violations or deviations were identified.

b. Audits

The NRC inspectors reviewed audits conducted by the onsite quality assurance department relevant to the station radiation protection program. Audit Report 35-81, "Radiation Exposure Documentation and Control," dated December 30, 1981, emanated from the audit conducted during the period December 9 through December 11, 1981. This audit was limited in scope to the thermoluminescent dosimeter (TLD); self-reading dosimeter (SRD); TLD and SRD calibration; and termination report documentation issued to individuals and the NRC. Audit 14-82 was conducted during the period March 24-26 and March 31 through April 2, 1982. This audit entitled, "Calibration Control," pertained to the instrumentation and control department functions and included all radiation protection instrumentation. Deficiencies identified during these audits, recommendations, and comments relating to the areas audited were corrected in a timely manner.

The NRC inspectors noted that one individual of the audit team held a senior reactor operator license and demonstrated some knowledge of the radiation protection program. The NRC inspectors discussed with the supervisor, quality assurance, the limited scope of previous audits. The licensee stated that future audits were being expanded to include all activities of the radiation protection program. Annual audits will include about one-third of the complete radiation protection program.

The NRC inspectors were unable to review audits conducted by the offsite Safety Audit and Review Committee or to determine the qualifications of audit personnel due to time limitations during this inspection. These audits will be reviewed during a future inspection.

No violations or deviations were identified.

c. Training

The NRC inspectors reviewed the Radiation Worker Training Program provided to employees, supplemental work force, and contractor-supplied personnel against the requirements of 10 CFR Part 19.12, "Instructions to Workers," and Regulatory Guides 8.27 and 8.29.

The NRC inspectors attended training courses presented for "Yellow Badge," "Escort," and "Blue Badge" access to the licensee's facilities. The licensee includes a "Practical Factors" training session as part of the "Blue Badge" requirement, in which personnel review the radiation work permit (RWP) requirements; properly suit up; remove all protective equipment; and perform a body frisk for radioactive contamination.

The NRC inspectors noted that all elements of Regulatory Guide 8.27 were not included in the licensee's training program; specifically bioassays, whole body counting, urinalysis, fecal analysis, and avoiding sample contamination. This is considered an open item (285/8226-01) and will be reviewed during a future inspection.

The NRC inspectors discussed with a licensee's representative the benefit of advising the radiation protection training instructors of personnel contamination incidents and for requiring the practical factors training session to be included as part of the retraining program.

The NRC inspectors verified that the training and retraining records of eight station employees and seven contractor-supplied personnel were complete.

No violations or deviations were identified.

d. Procedures

The NRC inspectors reviewed the licensee's procedures that had been revised during 1982 to determine compliance with 10 CFR Part 20 requirements and recommendations contained in Regulatory Guides 1.33, 8.8, 8.9, 8.15, 8.25; industry standards ANSI N13.1-1969, N13.11(draft), N13.12(draft), N18.1-1971, N18.7-1976, N322-1977, N323-1978, and N343-1978; and NUREG-0761.

The following procedures were reviewed during this inspection.

- RPP-18, "Instrument Selection," Revision 2, dated December 3, 1981.
- RPP-20, "Radiation Work Permits" (RWP), Revision 0, dated March 11, 1982.
- RPP-21, "Use of Extremity TLD's and Extremity Direct Reading Dosimeters," Revision 0, dated March 11, 1982.

No violations or deviations were identified.

e. Instruments and Equipment

The NRC inspectors reviewed the licensee's Radiation Protection Instrumentation Program to verify that monitoring instruments pertinent to worker safety are calibrated, operated properly, and agree with the recommendations of Regulatory Guides 8.4 and 8.25 and ANSI N13.1-1969 and N323-1978.

The NRC inspectors noted that the licensee did not have a portable alpha survey meter available to perform area or equipment surveys. A licensee's representative stated that two portable alpha survey meters were to be ordered and would be onsite prior to the next refueling outage scheduled for January 3, 1983. The NRC inspectors noted that the airborne radioactivity collection media (Staplex Filter TFA 2133) used by the licensee, did not meet ANSI-N13.1-1969 recommendations for alpha collection efficiency. The inspectors discussed with a licensee's representative the need to obtain a more suitable collection media for airborne radioactivity measurements. The inspectors also noted that the method used by the licensee to determine high volume air sampler air flow rates could result in significant errors. The licensee determines air flow rate by placing a clean filter in the high volume air sampler and measuring the air flow rate in a clean atmosphere. The recorded airflow is used for all air samples taken, no flow measuring device is used by the licensee. During operation, an increased pressure drop resulting from the ambient dust clogging on the filter will rapidly reduce the flow rate. The licensee makes no allowance for the decreased airflow when computing total air volume sampled. The licensee should reevaluate the method of calibration of air sample pumps recording airflow measurements and revise air sampling procedures. The licensee's representative stated that the air sampling program would be revised and an acceptable industry standard collection media would be in use by December 15, 1982, depending on availability. The inspectors stated that this will be considered an open item pending a complete revision to the air sampling program (285/8226-02).

The NRC inspectors were unable to complete a review of the licensee's calibration program of radiation protection instruments. The licensee stated that instrument records were offsite being microfilmed. This is considered an unresolved item pending an audit of instrument calibration frequency, calibration after maintenance, and full range calibration determinations (285/8226-03).

No violations or deviations were identified.

f. Exposure Control

(1) External Exposure Control

All personnel entering the radiation control area were routinely issued a TLD and an SRD. Additional dosimetric devices, such as high range SRD's or extremity TLD's, may be required in certain areas and are specified on the RWP.

The NRC inspectors reviewed the exposure records of 15 individuals selected at random from the licensee's exposure control records. These records showed that all individuals had current NRC Form 4 and 5's or equivalent as required by 10 CFR Parts 20.102 and 20.401.

(2) Internal Exposure Control

The NRC inspectors reviewed the licensee's internal exposure program to determine compliance with the requirements of 10 CFR Part 20.103 and the recommendations of ANSI N343-1978.

The internal exposure program consists of an initial whole body count (WBC), annual WBC for long-term employees, and an exit WBC for all OPPD employees and contractor personnel who enter radiologically controlled areas. Other bioassay sampling and counting are conducted when deemed appropriate by the health physics supervisor.

The NRC inspectors did not review the operation, daily checks, or calibration of the whole body counter. The station procedure for calibration of the WBC was in draft at the time of this inspection. This is considered an open item (285/8226-04) pending completion of the WBC calibration procedure and review against the recommendations of ANSI N343-1978.

The NRC inspectors did not review the licensee's respiratory protection program due to time limitations. This will be reviewed during a future inspection.

No violations or deviations were identified.

g. Posting, Labeling and Control

(1) Control

The NRC inspectors reviewed posting and control of radiologically controlled areas for compliance with 10 CFR Parts 20.105, 20.203, 20.207, and FCS Technical Specifications 5.11.1 and 5.11.2.

The NRC inspectors observed the licensee's control of contaminated tools and equipment used in the radiologically controlled areas. The licensee requires that tools and equipment be surveyed by radiation protection personnel prior to release to uncontrolled areas. Within the radiologically controlled area, several areas were used to store radioactive tools, equipment, and components. The inspectors noted that these storage areas were designated by a rope barrier with the appropriate radiation area sign, but provided poor control to prevent the addition or removal of materials to or from these areas.

(2) Posting of Notices

The NRC inspectors observed that the licensee had posted the necessary notices and information as required by 10 CFR Part 19.11.

No violations or deviations were identified.

h. Surveys

The NRC inspectors reviewed licensee radiation, contamination, and airborne radioactivity surveys to determine compliance with 10 CFR Parts 20.103, 20.201, and 20.401. Selected daily, weekly, and monthly routine radiation, contamination, and airborne surveys for the period January 16, 1982 through October 22, 1982, were examined. The inspectors made independent radiation measurements to verify recently reported licensee survey levels. The inspectors discussed airborne radioactivity surveys with a licensee's representative. See paragraph 4.e for details.

No violations or deviations were identified.

i. Notifications and Reports

The NRC inspectors reviewed the licensee's radiation exposure records and reports provided to individuals and to the NRC pursuant to requirements of 10 CFR Parts 19.13, 20.407, 20.408, and 20.409. The inspectors did not identify any errors or omissions involving termination reports.

No violations or deviations were identified.

5. Transportation Activities

The NRC inspectors reviewed the licensee's program for the receipt and shipment of radioactive and fissile materials to determine compliance with 10 CFR Parts 20 and 71; 49 CFR Parts 0-199; and the recommendations of Regulatory Guides 7.3 and 8.27; and draft NUREG-0731.

Documents Reviewed

- Dat-O-Line Manuals for 10 CFR Parts 20 and 71; and 49 CFR Parts 100 to 177
- Radioactive material receipt records for 1980, 1981, and 1982 (up to October 1982)
- Radioactive material shipment records for 1981 and 1982 (up to October 1982)
- FCS Operating Manual
- FCS Training Manual
- FCS Radiation Protection Manual (RPM)
- FCS Standing Order No. T-4, "Waste Solids Shipment," Revision 4, dated September 6, 1979
- FCS Instructor Lesson Plan, "General Employee Training - Blue Badge - Rad. Waste Reduction Program/Equipment Decon.," dated September 25, 1982
- OPPD Audit Report 30-82, "Radioactive Waste Packaging and Transfer," dated July 20, 1982
- Radiation Protection Procedure (RPP)-14, "In-Plant Collection and Disposal of Radioactive Waste," Revision 0, dated October 21, 1980
- RPP-15, "Verification of Liquid Free Solidified Waste, Spent Resin Waste, and Dry Compactible or Noncompactible Waste," Revision 2, dated January 29, 1981
- Health Physics (HP) Procedure 3, "Solid Waste Shipment Procedures," Revision 8, dated September 2, 1982
- HP-8, "Labeling and Bagging of Radioactive Materials," Revision 1, dated December 5, 1981
- Operating Instruction (OI) - Waste Disposal System (WDS)-2, "Solids Waste Disposal," Revision 8, dated January 19, 1982

- OI-WDS-5, "Backup Solidification Processing of Waste Concentrates," Revision 7, dated July 28, 1981
- Position Description, "Supervisor - Chemical and Radiation Protection," dated April 1982
- Position Description, "Plant Health Physicist," dated April 1982
- Position Description, "Chemistry/Radiation Protection Senior Technician," dated August 10, 1981
- Position Description, "Chemist," dated August 20, 1979

a. Management Controls

The NRC inspectors determined that Fort Calhoun Station (FCS) had identified in Standing Order T-4 (SO T-4) that both the supervisor - chemistry and radiation protection group and the plant health physicist (PHP) were responsible for ensuring offsite shipments of radioactive waste are conducted properly. The NRC inspectors found that the responsibility for performing the functional aspects of shipping station radioactive material was unofficially assigned to a technician in the CARP group. The technician's duties involving transportation activities were not found to be described in his current job description. The technician performed duties involving waste processing and transportation activities on a full-time basis.

The NRC inspectors determined also that the licensee had established detailed, though somewhat limited in scope, instructions for the conduct of radioactive material transportation, receipt, and packaging operations in station-approved procedures. These procedures were found in Section 6, "Radioactive Material Control," of the FCS Radiation Protection Manual, and procedures HPP-3, HPP-8, RPP-14, and RPP-15. Discussions and identification of deficiencies concerning FCS procedures are found in paragraph 5.1 of this report. Basically, the NRC inspectors found that FCS procedures for transportation activities cover adequately only the area concerning shipment of radioactive materials categorized as low specific activity (LSA) via sole-use-carrier (full load). The NRC inspectors determined that the licensee had previously performed shipments in other categories, such as limited quantities via common carrier-nonsole-use, LSA greater than Type A quantities, Type B quantities, spent fuel, and large quantities, and the licensee is expected to perform similar shipments in the future.

The NRC inspectors found the FCS procedures for transportation activities contained very little quality control activities to ensure all aspects of procedural requirements of NRC and DOT requirements had been met prior to offsite shipment of packages.

Furthermore, the NRC inspectors noted a definite lack of supervisory attentions to the day-to-day activities of the transportation and waste processing group. In regard to the aforementioned, it was determined that during a shipment of spent resin in 1981, which involved DOT and NRC regulations not normally encountered by the licensee, the required (SO T-4) CARP supervisory signatures on transportation records, indicating a responsible review of the shipment had been made, were delegated to other nonsupervisory HP personnel. This condition is not conducive to insuring transportation is properly carried out.

This item (285/8226-05) is considered open pending licensee action to:

- Provide a job description for the CARP technician performing waste processing and transportation activities that is more closely aligned with his actual duties.
- Implement procedures that provide detailed instructions for all aspects of station transportation activities and sufficient quality assurance activities/steps within the procedures to ensure all NRC and DOT requirements are complied with.
- Evaluate implementation of a periodic CARP supervisory audit/work operations review of transportation activities between management QA audits.

b. Selection of Packagings

The licensee did not have any procedures specifying requirements for the selection, procurement, fabrication, or reuse of appropriate packagings for shipping radioactive materials. The 1982 QA audit identified the need for FCS to include 55-gallon (DOT 17H) containers under the quality control procurement program. The NRC inspectors note that existing procedures do not provide sufficient instruction on selection of packaging for other than LSA shipments (strong tight container criteria) via sole-use-carrier. The licensee's shipment records appear to indicate that prior shipments of "limited quantities" and "LSA" shipments were properly performed and quantity limits were properly applied. The NRC inspector determined that the licensee maintained current documents regarding NRC certification of casks used and manufacturer's maintenance procedures for casks normally used by the licensee. A review of licensee shipping records and discussions with licensee's representatives indicated that the manufacturer's maintenance and loading procedures were normally used to prepare casks for shipment. The NRC inspector noted that licensee procedures do not reference the use of any specific cask maintenance or loading procedures to ensure that the applicable requirements of 10 CFR Parts 71.53 and 71.54 are complied with.

The licensee had only two shipments within the last 20 months that required special package consideration. One involved a dewatered resin shipment and the other a spent fuel shipment of which both were arranged and organized by a contractor. Both shipments appeared to meet all NRC and DOT requirements.

This item (285/8226-06) is considered open pending licensee action to:

- Provide suitable quality assurance procedures for the selection and qualification of packages used for the shipment of radioactive materials.

- Provide for implementation of special procedures for maintenance and loading of special shipping packages.

c. Preparation of Packages for Shipment

The NRC inspectors determined by review of licensee's shipment records and transportation activity procedures that very little instruction is provided for ensuring that any special packaging used for transportation of radioactive material meets the specified quantity of design and construction (49 CFR Part 173.393(m)(2)) and that manufacturer preuse inspection and preventive maintenance procedures are implemented.

The licensee's records indicated that the only radioactive liquids shipped by the licensee were small volume (approximately 4 liter) samples of reactor coolant and secondary system water shipped to an offsite analytical laboratory. These samples all were shipped as limited quantities and appeared to be properly classified in regard to the requirements of 49 CFR Part 173.391(a). The NRC inspectors are concerned that approved station procedures were not used in making these type shipments since they comprise approximately 10 percent of all licensee shipments.

During the NRC inspection, the licensee did not prepare any packages for shipment and only received radioactive sources in quantities not greater than Type A quantities.

The licensee's procedures appear to provide for the proper labeling and marking of shipping packages.

The licensee had provided detailed instructions in station procedures for monitoring of packages for radiation and radioactive contamination, and ensuring shipment met applicable NRC and DOT requirements as to surface contamination and radiation levels.

The licensee complied with the heat limit requirements of 49 CFR Part 173.393(c) during the irradiated fuel shipment in 1982.

The NRC inspectors determined by discussion with a licensee representative that QC activities for transportation activities were normally performed after the fact (after completion of the packaging/ shipment preparation) and consist mainly of a document review or as directed by the HP technician in charge of transportation activities. The licensee's procedures for transportation activities do not provide for independent QC activities or QC signoffs during packaging or package loading to verify that critical items have been accomplished properly. The NRC inspectors noted that there appeared to be sufficient quality control over ensuring that free liquids did not exceed burial contract specifications. The NRC inspectors noted that two checkoff lists were used for a Type B quantity shipment (81-08) of resin in an NRC certified cask. Both of the aforementioned checkoff lists were completed by the technician assigned responsibility for ensuring that station transportation activities are properly performed. The only place that independent QC personnel are specifically required is noted in procedure HPP-3 and only after the shipment is ready to leave the station.

This item (285/8226-07) is considered open pending licensee action to:

- Provide suitable independent quality assurance activities in radioactive material transportation procedures to ensure transportation activities are properly performed.
- Evaluate the use of a shipment specific (Type A or B quantities, spent fuel-large quantity, LSA, exempt, etc.) checkoff list for use by station QC personnel during inspection of transportation activities.

d. Delivery of Completed Packages to Carriers

The licensee did not ship any radioactive material offsite during this inspection, so a review of licensee's shipment records was conducted on records for the years 1981 and 1982, as noted in the "Documents Reviewed" section (5) of this report.

The licensee's shipping records appeared orderly and normally contained the following documents. The licensee provided this documentation (two sets) to the shipment carrier also.

- H.P. Waste Shipment Checkoff Sheet
- Form FC-218, "Radioactive Shipment Authorization"
- Various radiation and contamination survey sheets
- Form FC-203, "Specific Instructions for Maintenance of Exclusive Use Shipments Controls"

- South Carolina or Washington State Radioactive Waste Shipment Certification form

- South Carolina or Washington State Radioactive Waste Shipment Prior Notification and Manifest form

- Straight Bill of Lading form

- Radioactive Shipment Record form

The NRC inspectors noted that the licensee had not developed a procedure for the implementation of the prior notification of states and the NRC for shipments of radioactive material greater than the quantities specific in 10 CFR Part 71.5b. The licensee representative responsible for transportation activities was not familiar with the guidance provided in NUREG-0923, "Advance Notification of Shipments of Nuclear Waste and Spent Fuel." Even though the licensee had made a shipment of irradiated fuel in October 1982, the licensee depended on a contractor to make the needed notifications. The licensee's representative for transportation activities signed off on the "H.P. Waste Shipment Checkoff Sheet," that the required notifications had been made. A spot check of states involved in the transportation route verified notification, as well as verification of USNRC Region IV notification. No other licensee shipments required advance notification.

This item (285/8226-09) is considered open pending licensee action to:

- Implement procedures to assure compliance with the prior notification requirements specified in 10 CFR Part 71.5(b).

e. Loading of Packages on Transport Vehicles

Even though detailed procedures are not provided for all aspects of transportation activities, the licensee appears to have met all requirements for radiation levels, placarding (LSA only), removable surface contamination, blocking, and bracing for past radioactive material shipments. Procedures existed for notification of consignee, prior to shipment, of shipment dates, and special unloading instructions.

No violations or deviations were identified.

f. Receipt of Radioactive Material

The NRC inspectors reviewed licensee records and procedures governing the receipt of radioactive materials and determined that the licensee's program appeared satisfactory. The licensee's procedures provided for the implementation of the guidance provided in the NRC Regulatory Guide 7.3, "Procedures for Picking Up and Receiving

Packages of Radioactive Material." Records reviewed did not indicate that the licensee had received any shipments of radioactive materials in quantities greater than Type A quantities.

No violations or deviations were identified.

g. Incident Reporting

The NRC inspectors determined that the licensee did not act as a private carrier of radioactive material and the requirements imposed by 49 CFR Parts 171.15 and 171.16 were not applicable.

No violations or deviations were identified.

h. Indoctrination and Training Program

The licensee's training program for radioactive waste reduction, solid waste packaging, and liquid waste solidification in concrete and transportation requirements is accomplished mainly by self-study, on-the-job training, and supervisor review of employees' functional ability to perform specific jobs. The technician performing transportation activities appears to be the only CARP group member to have had specialized training in transportation regulations. This specialized training was provided by an offsite contractor and covered burial site requirements and NRC and DOT regulations governing radioactive material transportation. The technician appeared to be quite knowledgeable in transportation requirements and regulations.

The transportation technician also provides FCS personnel retraining in transportation activities via annual lectures. The transportation technician maintains his proficiency by self-study of industry periodicals subscribed to by the licensee and maintains updated versions of NRC and DOT code of federal regulations applicable to transportation of radioactive material.

The NRC inspectors determined by discussions with licensee personnel and a review of training records that personnel were provided licensee specified training in FCS transportation activities, and yearly retraining via lecture appears to be adequate.

The NRC inspectors noted to the licensee, with concern, that due to the high incidence of damaged containers occurring during solid waste compaction operations there may be a problem with employee training or procedure compliance. The inspectors determined that a nontrained contractor employee was performing solid waste compaction operations. The licensee was aware of this and indicated that nondocumented training, OJT, was given the contractor employee.

This item (285/8226-08) is considered open pending licensee action to:

- Evaluate training provided operators of the solid waste compaction equipment regarding instructions provided in station procedure OI-WDS-2, "Solids Waste Disposal."

i. Licensee's Audit Program

The NRC inspectors reviewed the licensee's QA audits of the FCS transportation activities. The OPPD QA Department had performed audits in 1981 and 1982 of "Radioactive Waste Packaging and Transfer." The NRC found the audits weak in that the audits failed to adequately verify licensee compliance with DOT and NRC regulations and licensee procedures. QA Audit Report 30-82 covered approximately 112 individual items and the NRC inspectors determined the following from a review of the audit:

- Only three items were directed at waste compaction or solidification activities and none of them involved direct observation of transportation operations.

- Audit item 8, required a response to the question, "Do all containers leaving the Fort Calhoun Station meet DOT requirements for shipping?" The answer was noted as "yes" with the following remark, "17H drums used." The NRC inspectors noted that the licensee uses containers other than DOT 17H drums.

- Audit item 27(a) required a response to "Container must meet DOT specification 7A requirements as listed in CFR 49 173.395a.1-4 and could be 30 gallon or 55 gallon size." The answer to this question was noted as "yes" with the following remark, "Verified only 55 gal. 17H drums used." The licensee could not provide the NRC inspector with the necessary certification, as required by 49 CFR Part 173.395(a)(1), that the licensee's 17H drums met or exceeded DOT specification 7A requirements. Note: The above is only used to point out deficiencies in the QA audit program and is not considered a violation of NRC or DOT requirements since a review of shipment records did not disclose any violations due to imposed requirements on specific shipping containers. None of the licensee's radioactive material shipments appeared to require use of DOT 7A specification containers.

The above deficiencies in the audit program appear to point out the need for a member of the audit team to be knowledgeable in the area of DOT and NRC transportation requirements.

This item (285/8226-10) is considered open pending licensee action to:

- Evaluate the special expertise needed for personnel performing future audits of FCS radioactive material transportation activities.

j. Examination of Licensee's Packages

The NRC inspectors toured licensee facilities used for the processing of radioactive wastes and the storage of packaged wastes. The NRC inspectors inspected the contents of a 128-cubic-foot wooden box used for the disposal of noncompactible LSA radioactive wastes. The NRC inspectors noted with concern the fact that the box contained three damaged and empty 55-gallon metal drums (DOT 17H). See paragraph 5.i discussion on waste compaction training. The NRC inspectors also observed a prestaged waste barrel for radioactive material full of assorted nonradioactive materials, such as old file folders, old maintenance orders, clear plastic file covers, and empty boric acid bags. The NRC inspectors pointed out to the licensee that these findings did not indicate the licensee had a very strong radioactive waste reduction program.

The NRC inspectors had the licensee open 3 waste drums, picked at [illegible] from 30 or more stored drums within the temporary storage [illegible], located within the new fuel inspection room. Two of the drums contained solidified liquid wastes in concrete; no visible standing liquids were observed or indicated. The concrete appeared to be of good quality.

One drum contained compacted wastes of plastics, cloth, and paper with plastic compaction discs used within the drum as an aid in achieving greater density compaction. All drums appeared to be of sound condition and properly sealed with the use of a gasketed lid.

The NRC inspectors noted that due to the quality of post filling free standing liquid inspections for all waste drums there is minimal chance that a waste drum leaves the FCS with free standing liquids in excess of burial site contract requirements. The licensee had temporarily stored, in 3 different areas of the Auxiliary Building, a total of approximately 200 cubic feet of used structural wood (left over from a previous station outage), and approximately 70 or more damaged and undamaged 55-gallon containers filled with oils, other liquids, compactible wastes, or solidified liquid wastes. The NRC inspectors also noted that the licensee had on hand several six-foot tall demineralizers, and concrete-encased liquid waste processing system filters. The licensee's representative indicated that all the 55-gallon drums are awaiting reprocessing, the six-foot tall demineralizers (they were generated by a contractor during clean up of nonradiological contamination of the spent fuel pool water), and are being shipped out one or two at a time with each waste shipment, and the concrete-encased filters are awaiting technical instructions for determining the correct curie content or until another batch of filters is processed so that the needed data is collected. The licensee did not maintain an official inventory of accumulated radioactive materials/waste in long-term storage or waste in need of processing.

The NRC inspectors noted with concern the number, approximately 10-15 drums, of 55 gallon containers that were damaged during waste compaction operations. Some of the drums were released from the waste compaction area with significant damage. The NRC inspectors also noted that FCS is scheduled for a maintenance outage in early 1983 which will further tax the available storage spaces already in use. See discussion of operator training in paragraph 5.h of this report.

The NRC inspectors noted that the licensee had installed a differential pressure indicator across the high efficiency filter on the waste compactor and instructions for determining when filter must be evaluated for replacement in the form of a sign next to the differential pressure gauge. This item was identified as an open item in NRC Inspection Report 50-285/80-04. The licensee demonstrated the differential pressure gauge operability.

This item (285/8226-11) is considered open pending licensee action to:

- Investigate and take action to reduce the damage to 55-gallon containers used for compaction of solid wastes and improve the preuse inspection of containers.

- Reduce the amount of nonradioactive waste being disposed of as radioactive waste.

- Implement procedures that will ensure that wastes stored for later processing are maintained on an inventoried status and periodically actions are taken to evaluate reduction of packaged and unpackaged wastes awaiting processing or repackaging

k. Radioactive Material Receipt and Shipment Records

The NRC inspectors reviewed licensee's records for the receipt and shipment of radioactive materials. Licensee's records involving the receipt of radioactive materials and their continued accountability appear to be satisfactory. The licensee had not appeared to have received any radioactive materials in excess of Type A quantities and the majority of the incoming shipments involved near-exempt quantities as identified in 10 CFR Part 20.205.

The licensee's records for the transportation of radioactive materials during 1981 and 1982 were reviewed. The NRC inspector noted that the licensee has made 18 shipments of radioactive material during the first half of 1982 which is the same as the total number of shipments made in all of 1981. The licensee's shipments consisted mainly of radioactive wastes to either Barwell, South Carolina or Handford, Washington, low-level waste repositories. The licensee maintains current copies of contracts with each of the above-noted burial sites.

The NRC inspectors noted that shipment records, especially FCS Form FC-218, did not provide sufficient detail to ensure all aspects of labeling, identifying, and packaging required by NRC and DOT regulations are adequately addressed. Licensee representatives acknowledged that due to the multitude of regulatory requirements involving transportation of radioactive materials, i.e., hazardous materials per DOT regulations, an evaluation of shipping documents would be conducted for possible improvement.

This item (285/8226-12) is considered <u>open</u> pending licensee action to:

- Determine the effectiveness of the licensee's radioactive shipment record system to provide verification of compliance with NRC and DOT transportation regulations.

l. <u>Licensee Solid Waste Packaging and Transportation Activity Procedures</u>

The NRC inspectors reviewed the licensee's procedures for the packaging and shipment of radioactive materials. These comments are in addition to the findings discussed in other sections of this report on the lack of specific procedures for other aspects of radioactive material transportation.

(1) Radiation Protection Manual, Section 6.0, "Radioactive Material Control," Revision 11, dated July 7, 1982.

(a) Paragraph 6.3.4.3 does not provide necessary instructions to ensure that packages selected are accompanied by documented certification that the package meets or exceeds 49 CFR Part 173.398 requirements.

(b) Paragraph 6.3.45 does not provide suitable detail to ensure that Type A quantities or large quantities are properly packaged to comply with the DOT requirements specified in 49 CFR Part 173.393(g).

(c) Paragraph 6.4.1 contains conflicting and unclear limits on removable surface contamination limits for packages/containers.

(d) Paragraph 6.4.1 references the use of a conversion factor of 0.3 for determination of the curie content of a 55-gallon drum of waste. This item was previously reported as an open item in NRC Inspection Report 50-285/80-16. The NRC inspectors determined that this conversion factor does not provide adequate estimation of the curie content of 55-gallon drums with densities of filler material varying between .6 grams per cubic centimeter (g/cc) and 1.6 g/cc when estimated using accepted health physics practices.* The NRC inspectors also noted that this paragraph recognized waste as only being packaged as LSA in 55-gallon containers when the licensee uses other packages, such as the 128-ft^3 box. The licensee was found to be using an unverified conversion factor for computation of the curie content of this package also. The licensee needs to provide concise and comprehensive curie content estimation instructions to all commonly used station packages and quality control checks to ensure estimation made outside the scope of the instructions are accurate and formulated in accordance with industry standards or accepted practices.

(e) Paragraph 6.5 and Form FC-218 do not provide for the specifying of the DOT hazardous material identification number as required by HPP-3 and DOT regulations.

Note:

Generally, each licensee procedure that references just Title 49 provides very little useful information on specific regulatory requirements and methods to employ in complying with them for various types of radioactive material shipments.

(2) RPP-14, "In-Plant Collection and Disposal of Radioactive Waste," Revision 0, dated October 21, 1980.

. The NRC inspectors had no comment on this procedure.

(3) RPP-15, "Verification of Liquid Free Solidified Waste, Spent Resin Waste, and Dry Compactible or Noncompactible Waste," Revision 2, dated January 29, 1981.

. The NRC inspectors had no comment on this procedure.

(4) HPP-3, "Solid Waste Shipment Procedures," Revision 8, dated September 2, 1982.

(a) Paragraph E.3 of this instruction is too general in nature to ensure compliance with NRC and DOT regulations governing other than LSA shipments via sole-use-carrier.

(b) Paragraph E.7 - same comment as (a) above.

*W. B. Bowman II and D. L. Swindle, "Determination of the Ci Content of Packaged Radioactive Waste Using Measured Dose Rates," Health Physics 31, pp. 445-450 (1976).

(c) Paragraph E.8 instructions do not provide sufficient instructions for the marking and labeling of waste containers other than drums. Examples: 128-ft^3 wood box and 6-foot tall demineralizers are frequently shipped by the licensee.

(d) Paragraph E.11 instructions for determining waste container curie content are covered in the NRC inspectors' comments on Section 6.0 of the Radiation Protection Manual. See paragraph 5.1(1)(d) for details.

(e) Paragraph E.15 does not provide sufficient instructions to ensure that casks are properly inspected prior to use, that specific manufacturer procedures are followed for loading or unloading, or that the cask is properly closed and sealed. Example: No provisions for inspecting a contractor cask prior to use, as in verifying that the cask drain plug (if so equipped) is properly installed and lock wired in the sealed position.

(f) Paragraph E.22 does not provide for the licensee to verify by independent quality control inspection that special containers purchased by the licensee meet burial contractor specifications.

(g) Paragraph E.11 does not ensure that shipments of radioactive wastes or other materials are conducted properly since QC personnel are not required until all physical operations are completed, such as those involving the loading of individual packages, verification of package content, loading/closing of cask or vehicle, and completion of paper work.

(5) OI-WDS-2, "Solids Waste Disposal," Revision 8, dated September 19, 1982.

(a) Paragraph II.A should include instructions on verifying that the high efficiency filtered ventilation system on the waste baler is operating properly, including the differential pressure gauge which should be checked for both minimum and maximum values when initially put in service and during operation.

(b) This procedure does not provide instructions for properly accomplishing the following tasks performed by the licensee during waste compaction.

- Insertion of the plastic discs into drums of compacted waste to assist in achieving high density compactions.

. Closure of the filled drums.

. Weighing and marking of weight on drums.

This item (285/8226-13) is considered open pending licensee actions to:

. Resolve NRC inspectors' comments to the above cited procedures.

6. NUREG-0737, "Clarification of TMI Action Plan Requirements"

The NRC inspectors reviewed the licensee's programs and commitment in meeting the post-TMI requirements according to NUREG-0737 for:

Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operation."

Item II.B.3, "Postaccident Sampling Capability."

Item II.E.4.2, "Containment Isolation Dependability, Position (7), Containment Purge and Vent Isolation Valves Must Close on a High Radiation Signal."

Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 1, "Noble Gas Effluent Monitor," Attachment 2, "Sampling and Analysis of Plant Effluents," and Attachment 3, "Containment High-Range Radiation Monitor."

Item III.D.3.3, "Improved Inplant Iodine Instrumentation Under Accident Conditions."

Item III.D.3.4, "Control-Room Habitability Requirements."

a. Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operation"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to All Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 25, 1979, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(c) Letter, October 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, November 27, 1979, to H. R. Denton (USNRC) from W. C. Jones (OPPD)

(e) Letter, December 21, 1979, to H. R. Denton (USNRC) from W. C. Jones (OPPD)

(f) Letter, April 7, 1980, to W. C. Jones (OPPD) from R. W. Reid (USNRC)

(g) Letter, October 6, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(h) Letter, December 12, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(i) Letter, December 31, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(j) Letter, February 27, 1981, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(k) Letter, July 10, 1981, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(l) Letter, December 28, 1981, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(m) Letter, March 17, 1982, to All Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(n) Letter, April 1, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(o) Letter, May 5, 1982, to All Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(p) Letter, April 30, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(q) Letter, June 8, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(r) Combustion Engineering Report, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces and Systems Which May Be Used in Post-Accident Operations (NUREG-0578, Section 2.1.6.6) for Fort Calhoun Station, Omaha Public Power District," December 1979, CE-18074-648

(s) Combustion Engineering Report, "Post Accident Radiation Level, Final Report," CE-18074-998, July 15, 1981

(t) Standard Review Plan 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Features Components Outside Containment"

(u) Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants 19 - Control Room."

(v) Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant for Pressurized Water Reactors"

(w) Memorandum, September 1, 1982, to R. A. Clark, Chief Operating Reactors Branch Number 3 from E. Tourigny, Lead PM, Plant Shielding Modifications.

(2) Discussion

NUREG-0737 for this item (Item II.B.2) states that with the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guide 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station, and sample analysis area must be included among those areas where access is considered vital after an accident. The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

The design dose rate for personnel in a vital area should be such that the guidelines of GDC 19 will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, NUREG-0737 provides the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident.

The control room and onsite technical support center are areas that will require full-time occupancy (<15 mrem/hr averaged over 30 days) during the course of the accident.

(3) <u>Conclusions</u>

The above referenced design review of plant shielding study (see 6.a(8)) reviewed the following systems and vital areas:

- Safety injection system
- Containment spray system
- Emergency diesel generators
- Station batteries

- Chemical and volume control charging system
- Component cooling water system
- Reactor coolant and containment atmosphere sample system
- Radioactive waste disposal system
- Shut down cooling system
- Chemical and volume control letdown system
- Onsite technical support center
- Control room complex
- Diesel generator rooms
- Storage battery rooms
- Radioactive waste disposal system control panel area
- Reactor coolant and containment atmosphere sampling room
- Radiochemistry and chemical analysis laboratory
- Mechanical penetration room
- Motor control center
- Personnel corridor, which provides access to the systems and spaces defined above as being vital to postaccident recovery operations

The design review study was performed to meet the criteria given in Regulatory Guides 1.4, 1.7, and the NRC Standard Review Plan 15.6.5.

The results of this study show the dose equivalent rate to be approximately 9 mrem/h in the control room. This meets the General Design Criteria 19, which stipulates that the dose equivalent rate must be less than 15 mrem/h for continuous occupancy.

As a result of the design review study, four modifications were made as follows:

- Concrete shield wall installed on west side of SW corner of corridor serving control room complex

- Concrete shield wall installed on north wall of Pipe Penetration Area (shields personnel area corridor, Radio Chemistry and Chemical Analysis Laboratories)

- Concrete shield wall installed around Radwaste Control Area

- Installation of the remote level sensors in the sumps of the Safety Injection and Containment Spray Pump Areas 1 and 2 which read out in the control room

The review responsibility of Radiation Qualification of Safety-Related Equipment portion of this item (Item II.B.2) is no longer a responsibility of the USNRC Regions (see enclosure 4 of paragraph 6.a(1)(w))

It was concluded during this inspection that this item (Item II.B.2) for Fort Calhoun Nuclear Plant met the NUREG-0737 criteria and is, therefore, considered acceptable and closed.

No violations or deviations were identified.

b. Item II.E.4.2, "Containment Isolation Dependability"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to All Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 25, 1979, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(c) Letter, October 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Memorandum, January 5, 1980, to J. P. O'Reilly (USNRC) from T. V. Donat (USNRC)

(e) Letter, March 13, 1980, to R. W. Reid (USNRC) from W. C. Jones (OPPD)

(f) Letter, April 7, 1980, to W. C. Jones (OPPD) from R. W. Reid (USNRC)

(g) Letter, October 6, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(h) Letter, July 10, 1981, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(i) Letter, February 12, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(j) Letter, March 17, 1982, to All Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(2) Discussion

According to NUREG-0737, the containment purge and vent isolation valves must close on a high radiation signal and the sealed-closed purge isolation valves shall be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 24 hours that the purge valves are closed.

(3) Conclusions

The NRC inspectors reviewed the Fort Calhoun Station Technical Specifications (Tables 2.1 and 2.4) and Updated Safety Analysis Report (Section 7.3.2.6 and Table 5.9). These verify that a ventilation isolation actuation signal initiates automatic closing of the pertinent containment purge and vent isolation valves, if open, except for the H_2 purge isolation valves which are manually operated and are normally locked closed.

Therefore, it is concluded that this item (Item II.E.4.2) met the NUREG-0737 criteria and is considered acceptable and closed.

No violations or deviations were identified.

c. Item II.F.1, "Additional Accident Monitoring Instrumentation"

(1) Attachment 3, "Containment High-Range Radiation Monitor"

(a) Documents Reviewed

i. Letter, September 13, 1979, to All Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

ii. Letter, October 25, 1979, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

iii. Letter, October 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

iv. Letter, November 27, 1979, to H. R. Denton (USNRC) from W. C. Jones (OPPD)

v. Memorandum, January 5, 1980, to J. P. O'Reilly (USNRC) from T. J. Donat (USNRC)

vi. Letter, April 7, 1980, to W. C. Jones (OPPD) from R. W. Reid (USNRC)

vii. Letter, October 6, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

viii. Note, July 22, 1981, to R. J. Mattson (USNRC) from W. E. Kreger (USNRC)

ix. Memorandum, December 23, 1981, to T. M. Novak (USNRC) from W. E. Kreger (USNRC)

x. Letter, December 28, 1981, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

xi. Letter, March 19, 1982, to All Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

xii. Letter, April 21, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

xiii. Letter, April 30, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

xiv. Gibson Hill Drawing 11405 E 94, "Building Containment Tray and Conduit Lay-out Plan," Operating Plan at 1045'-0"

xv. Gibson Hill Drawing 11405-A-14, "Primary Plant," Section B-B

xvi. Letter, October 28, 1980, to L. Sojaha (OPPD) from S. R. Pressman (Victoreen, Inc.)

xvii. Omaha Public Power District MR-FC-79-190, "Containment High Range Monitor"

xviii. Fort Calhoun Station, "Containment High Range Monitor Calibration Procedure," CP-RM-091 A & B

(b) Discussion

For the item (Item II.F.1, Attachment 3), NUREG-0737 requires the licensee to provide two radiation monitor systems in containment with the capability to detect and measure the radiation level within the reactor containment during and following an accident.

In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979, letter to provide for a photon-only measurement with an upper range of E+07 R/hr.

The monitors shall be located in containment(s) in a manner so as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

The monitors are required to respond to gamma photons with energies as low as 60 keV to 3 MeV photons, with linear energy response ± 20 percent for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information. Monitors that use thick shielding to increase the upper range will underestimate postaccident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.

The licensee is required to calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to E+06 R/hr. Prior to initial use, the licensee must certify calibration of each detector for at least one point per decade of range between 1 R/hr and E+03 R/hr.

(c) Conclusions

The NRC inspectors determined that the Victoreen Radiation Detector (Model 877) of the Victoreen High Range Containment Monitor System (Model 875) has a range of E+07 R/hr for gamma radiation. These detectors are read out on a control room panel which includes high radiation, alert, failure alarms, indicating lights, and recorder. This system is powered by redundant vital instrument buses.

The containment High-Range Radiation Monitor (Model 877) will respond to within ± 20 percent from 80 keV through 3 MeV, and detects 60 keV gamma radiation.

The detectors were determined to be functional in an accident environment. The generic qualification tests included temperature and pressure during LOCA simulation, relative humidity, chemical spray, irradiation, and qualified life.

The two monitors were located in containment approximately 180° apart on the steam generator's shield walls. They were placed about five feet six inches above the floor making their accessibility easy. Although the NRC inspectors believe the location of these monitors will provide a reasonable assessment of the containment radiation conditions during an accident situation, there is some concern that the monitors do not "view" the majority of the containment volume since the steam generator's shields "shadow" a large fraction of the central volume of the containment building.

Prior to initial use, the monitors were calibrated by Victoreen at 75, 210, and 1900 R/hr before installation. After installation, the licensee calibrated the monitors in situ with a Co-60 source at 12, 35, and 300 R/hr, and also electronically at 10, E+02, E+03, E+04, E+05, and E+06 R/hr.

These monitors were installed during the last refueling outage and are scheduled for recalibration during every refueling outage. NUREG-0737 specifies that the detectors must be calibrated with a calibrated radiation source for at least one decade below 10 R/hr. The present calibration procedures (CP-RM-091 A & B) do not include this, therefore, this is considered an <u>open</u> item (285/8226-17) pending a change in the calibration procedures to include calibration at one decade below 10 R/hr.

No violations or deviations were identified.

d. <u>Item III.D.3.3</u>, "Improved Inplant Iodine Instrumentation Under Accident Conditions"

(1) <u>Documents Reviewed</u>

(a) Letter, September 13, 1979, to All Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 25, 1979, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(c) Letter, October 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, November 27, 1979, to H. R. Denton (USNRC) from W. C. Jones (OPPD)

(e) Letter, March 13, 1980, to R. W. Reid (USNRC) from W. C. Jones (OPPD)

(f) Letter, April 7, 1980, to W. C. Jones (OPPD) from R. W. Reid (USNRC)

(g) Letter, October 6, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(h) Letter, December 31, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(i) Letter, July 10, 1981, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(j) Note, July 22, 1981, to R. J. Mattson (USNRC) from W. E. Kreger (USNRC)

(k) Letter, March 12, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(l) Memorandum, April 26, 1982, to T. M. Novak (USNRC) from R. W. Houston (USNRC)

(m) Letter, April 29, 1982, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(n) Radiation Protection Procedure-16, "Airborne Radioiodine Monitoring"

(o) Emergency Plan Implementing Procedures-EOF-3, "Emergency Instruments and Equipment"

(2) Discussion

This item (Item III.D.3.3) requires that each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Each licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer (SCA). The SCA window should be calibrated to the 365 KeV of iodine-131 using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

(3) Conclusions

The NRC inspectors inspected the Eberline SAM-2, Dual Channel Analyser and Eberline RD-22 detector, and a Radeco H809V air sampler. This equipment is located in the Technical Support Center (TSC).

The Eberline SAM-2 system is portable to the extent that a hand truck can be used to move the detector shield (lead pig). The Eberline SAM-2 system is calibrated on a 6-month schedule according to Calibrating Procedure, CP-SAM-2. The NRC inspectors observed from this procedure that the licensee was calibrating the analyser to the Ba-133 gamma of 356 KeV instead of the 365 KeV gamma radiation of I-131 as recommended in NUREG-0737.

The licensee uses a Gelman glass particulate filter, type A-E, in conjunction with a silver zeolite cartridge in the Radeco H809V air sampler. Since silver zeolite is used, there is no need to purge the cartridge before counting because no noble gases will be entrapped.

The NRC inspectors noted an area of concern that the licensee has only one air sampler dedicated to the sampling of iodine in an accident condition. The NRC inspectors expressed this concern upon several occasions, including the exit meeting, and recommended more air samplers be made available in case of an accident. This concern was also expressed in the memorandum of April 26, 1982 (see 6.d(1)). The licensee has four additional air samplers, but these are located in the Emergency Off-Site Facility (EOF) and are to be used for off-site air sampling and probably would not be available for inplant sampling in an accident situation. The licensee should ensure that a backup air sampler is available in case the dedicated sampler becomes inoperative during an accident or more than one sampler is needed. NUREG-0737 states, "There should be sufficient samplers to sample all vital areas."

The licensee conducts training in the operation of the analyser and air sampler for iodine determination for licensee and contractor personnel on an annual schedule and when qualifying for shift health physics technician, respectively.

The TSC contains a permanently installed Eberline Particulate, Iodine, Noble Gas (PING) monitor which is calibrated on a refueling outage schedule according to Calibration Procedure CP-RM-67.

It is concluded that this item (Item III.D.3.3) met the NUREG-0737 criteria and is acceptable and closed.

No violations or deviations were identified.

e. Item III.D.3.4, "Control-Room Habitability Requirements"

(1) Documents Reviewed

(a) Letter, September 13, 1979, to All Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(b) Letter, October 25, 1979, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(c) Letter, October 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

(d) Letter, April 7, 1980, to W. C. Jones (OPPD) from R. M. Reid (USNRC)

(e) Letter, June 17, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(f) Letter, October 6, 1980, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(g) Letter, January 26, 1981, to D. G. Eisenhut (USNRC) from W. C. Jones (OPPD)

(h) Letter, July 10, 1981, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(i) Note, July 22, 1981, to R. J. Mattson (USNRC) from W. E. Kreger (USNRC)

(j) Letter, August 7, 1981, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(k) Letter, November 19, 1981, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(l) Memorandum, December 21, 1981, to R. A. Clark (USNRC) from R. N. Houston (USNRC)

(m) Letter, December 30, 1981, to W. C. Jones (OPPD) from R. A. Clark (USNRC)

(n) Letter, January 8, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(o) Letter, May 5, 1982, to All Licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(p) Letter, June 4, 1982, to R. A. Clark (USNRC) from W. C. Jones (OPPD)

(2) Discussion

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. NUREG-0737 specifies that licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity
2.2.3	Evaluation of Potential Accidents;
6.4	Habitability Systems

shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

- Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release";

- Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and

- K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability, but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident (LOCA) containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems.

(3) <u>Conclusions</u>

Enclosure 2 of the January 26, 1981, letter (see 6.e(g)) gives the licensee's submittal of their control room habitability study. This study included the following:

- Data Required for Control Room Habitability Evaluation
- Site Building Layout and Meteorology
- Control Room Characteristics
- Control Room Modes of Operation
- On-site Storage of Hazardous Chemicals
- Off-site Manufacturing, Storage and Transportation Facilities of Hazardous Chemicals

- Technical Specification
- Assumptions and Initial Conditions
- Control Room Habitability Design Evaluation and Analysis
- Assessment of Hazardous Chemical Materials
- Worst Case Chemical Accident Release
- Chemical Transport Accident Probability
- Control Room Infiltration and Exfiltration Rates
- Design Basis Chemical Accident Consequences
- Assessment of Radiological Events
- Airborne Radiological Consequences
- Control Room Shielding Design Review
- Emergency Instrumentation and Procedure Review
- Control Room Sustained Occupancy Review

The existing Fort Calhoun Station Control Room envelope and habitability systems have been evaluated and analyzed to determine the adequacy with which Control Room operators are protected against the effects of an accidental release of either toxic or radioactive gas. This allows the nuclear power plant to be safely operated or shut down under design basis accident conditions when the following modifications (which are a result of the control room habitability study) are completed.

MODIFICATIONS	SCHEDULED COMPLETION
Instrumentation for detection of airborne iodine radiation in the control room	6/31/83
Electrical and mechanical modifications to the HVAC system	6/31/83
Instrumentation for monitoring of toxic chemical gases	1/1/84

The licensee informed the NRC of these modifications and their completion dates in a letter of August 7, 1981 (see 6(1)(j)).

When these modifications are completed, then the licensee will have met the NUREG-0737 criteria for control room habitability, but until then, this is considered an open item (285/8226-18) pending the completion of the above modifications.

No violations or deviations were identified.

7. The NRC inspectors were unable to review the following NUREG-0737 items:

Item II.B.3, "Postaccident Sampling Capability,"

Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 1, "Noble Gas Effluent Monitor," and Attachment 2, "Sampling and Analysis of Plant Effluents," because the licensee had not completed these items. Although these items were supposed to be in effect January 1, 1982, per NUREG-0737, the licensee has corresponded, upon several occasions, with the Office of Nuclear Reactor Regulation, informing them of the new operational dates. The latest operational date for these NUREG-0737 items is November 30, 1982. Items II.B.3, II.F.1, Attachments 1 and 2 are considered open (285/8226-14, 285/8226-15, and 285/8226-16, respectively) pending completion of these items.

8. Unresolved Items

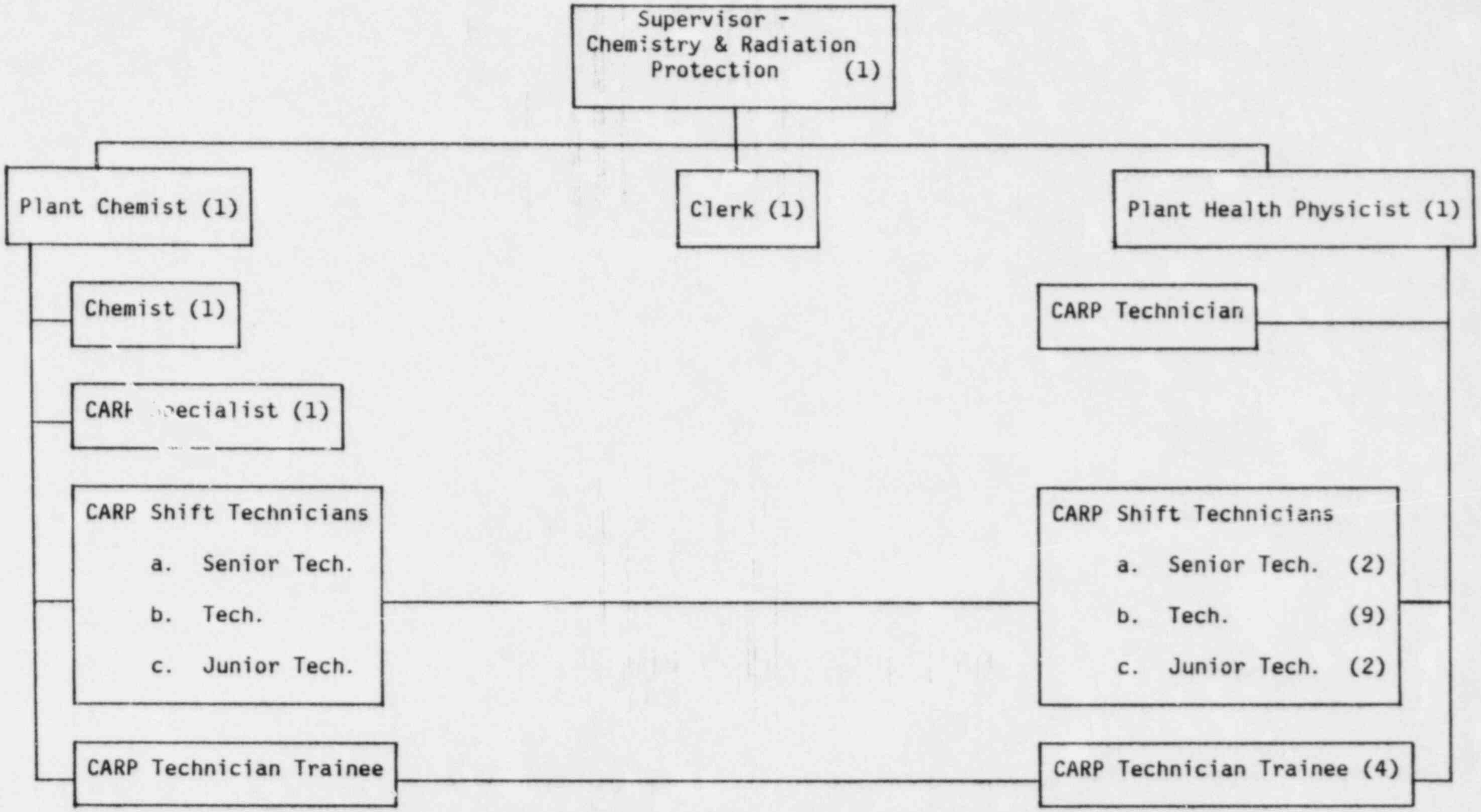
Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. (See paragraph 4.e).

9. Exit Interview

At the conclusion of the inspection on October 29, 1982, the NRC inspectors met with those persons identified in paragraph 1. The inspectors reviewed the scope of the inspection and the inspections findings. The licensee provided commitments for certain items discussed during the exit interview (see paragraph 4).

FIGURE 1

FCS - RADIATION PROTECTION ORGANIZATION



( ) Denotes number of individuals on duty.