APPENDIX C

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-298/82-32

Docket: 50-298 License: DPR-46

Licensee: Nebraska Public Power District (NPPD)
P.O. Box 499
Columbus, NE 68601

Facility Name: Cooper Nuclear Station (CNS)

Inspection At: Brownville, NE

Inspection Conducted: November 29-December 3, 1982

Inspectors: _____ 2/25/83
R. E. Baer, Radiation Specialist Date

for _____ 2/25/83
L. Wilborn, Radiation Specialist Date

_____ 2/25/83
W. Holley, Radiation Specialist Date

Approved: _____ 2/25/83
B. Murray, Chief Date
Facilities Radiation Protection Section

_____ 2/28/83
T. Westerman, Chief, Reactor Project Date
Section A

Inspection Summary

Inspection Conducted November 29-December 3, 1982 (Report 50-298/82-32)

Areas Inspected: Routine, unannounced inspection of the licensee's radiation protection program during operations including qualifications of personnel,

audits, training, radiological protection procedures, instruments and equipment, exposure control, posting and control, surveys, notifications, and reports; transportation activities including management controls, selection of packagings, preparation of packages for shipment, delivery of completed packages to carriers, receipt of packages, incident reporting, indoctrination and training, audits, examination of packages, and recording keeping; and selected NUREG-0737 items. The inspection involved 96 inspector-hours by three NRC inspectors.

Results: Within the 21 areas inspected, 2 violations and 1 deviation were identified (medical determination of respirator users - paragraph 4d and transportation of licensed material - paragraph 5i, and training records - paragraph 5d). Nine open items are discussed in paragraph 3.

Details

1. Persons Contacted

Principal Licensee Personnel

*L. Lessor, Station Superintendent
P. Borer, Operations Supervisor
J. Cline, Health Physics Secretary
W. Gilbert, Training Coordinator
R. Krause, Plant Engineer
J. Kutter, Lead Health Physics Technician
R. McDonald, Health Physicist
J. Mehser, Radwaste Operator
J. Sayer, Chemistry and Health Physics Supervisor
F. Shaw, Senior Site Mechanical Engineer
G. Smith, Quality Assurance Specialist
J. Warren, Chemist
V. Wolstenholm, Quality Assurance Superviosr

Other Personnel

D. Nizolek, Third Party Inspector, Nevada Inspection Service (State of Nevada Contractor)
D. DuBois, Senior Resident Reactor Inspector, USNRC

The inspectors also interviewed several other licensee employees including health physics and chemistry technicians, operators, and maintenance personnel.

*Denotes those present at the exit interview on December 3, 1982.

2. Licensee Action on Previous Inspection Findings

(Closed) Open Item (298/8108-01): Qualification Criteria for Health Physics Technicians - This item was discussed in NRC Inspection Report 50-298/81-08 and involved the lack of qualification criteria for health physics technicians. The licensee revised Station Procedure 1.5, "Selection and Training of Station Personnel," Revision 8, Section VI.3.D, to require that technicians in responsible positions shall have a minimum of 2 years of working experience in their specialty. This item is considered closed.

(Closed) Open Item (298/8108-02): Quality Control Checks on Vendor-Supplied Thermoluminescent Dosimeters (TLD) - This item was discussed in NRC Inspection Report 50-298/81-08 and involved the failure of the licensee to perform a quality control check on the vendor's performance by supplying TLD's irradiated to known doses. The licensee revised Station Procedure 9.1.1.3, "Personnel Dosimeter Program," Section VI.6, to include a

quality control check of vendor's performance on a quarterly basis. This item is considered closed.

(Closed) Open Item (298/8108-03): Whole Body Counter Action Levels - This item was discussed in NRC Inspection Report 50-298/81-08 and involved the whole body counter action level for nuclides routinely encountered at CNS. The licensee identified the potential intake of nuclides at the 40 maximum permissible concentration hour control measure level and revised Station Procedure 9.1.8, "Bio-Assay Whole Body Counting," Section VI.E.2. Whole body counter action levels were reduced from 10 percent to 5 percent of the maximum body burden recommended by the ICRP. This item is considered closed.

(Closed) Unresolved Item (298/7712-02): Stack Gas Monitors Inoperative - This item was discussed in NRC Inspection Report 50-298/77-12 and involved the stack gaseous effluent monitor (GE) being in the "purge" or bypass mode and a second stack monitor (NMC) not in operation. The licensee initiated a Nonconformance Report NCR001135 and verified that the NMC monitor was in operation during the time the GE monitor was out of service. This item is considered closed.

(Closed) Severity Level IV Violation (298/8220-03): Failure to Follow Procedures, Beta Counting of Air Samples - This item was identified in NRC Inspection Report 50-298/82-20 and involved the failure to count air sample filters on a beta counter as required by station procedure. The licensee revised Station Procedure 9.3.6.1, "Low Volume and High Volume Air Sampler Operation and Calibration," Revision 7, August 2, 1982, Section VI.C "note," which allows the licensee to perform a preliminary determination with a HP-210 probe for gross beta activity present on an air sampler filter. This item is considered closed.

(Closed) Severity Level IV Violation (298/8220-04): Failure to Follow Procedures, Extending the Expiration Date on Special Work Permits (SWP) Without Reevaluating Radiological Conditions - This item was identified in NRC Inspection Report 50-298/82-20 and involved extending the expiration date on a SWP beyond 1 calendar month without a reevaluation of the radiological conditions. The licensee revised Station Procedure 9.1.1.4, "Special Work Permits," to allow flexibility on expiration of SWP's on weekends, holidays, or high workload days. This item is considered closed.

(Closed) Severity Level IV Violation (298/8220-06): Failure to Post Radioactive Material Area - This item was identified in NRC Inspection Report 50-298/82-20 and involved the failure to post radioactive material storage areas within the licensee's protected area. The licensee has posted those areas where radioactive material was being stored. This item is considered closed.

(Closed) Severity Level IV Violation (298/8220-07): Failure to Perform Beta Radiation Surveys - This item was identified in NRC Inspection Report 50-298/82-20 and involved the failure to perform an evaluation of

the radiation hazards, specifically beta radiation surveys, in working areas. The licensee has revised Station Procedure 9.1.1.4, "Special Work Permit," requiring a beta radiation survey to be performed when smearable contamination levels exceed 22,000 disintegrations per minute per 100 square centimeters (dpm/100 CM^2). This item is considered closed.

(Closed) Severity IV Violation (298/8220-08): Failure to Document Radiation Surveys - This item was identified in NRC Inspection Report 50-298/82-20 and involved the failure to document nonroutine radiation surveys on CNS HP-100 data form as required by Station Procedure 9.2.1, "Radiation and Contamination Survey Frequency." The NRC inspectors verified that nonroutine radiation surveys were being documented as required on CNS HP-100 data forms. This item is considered closed.

(Open) Open Item (298/8220-01): Exposure Control - The NRC inspectors reviewed this item and determined the licensee still needs to finalize procedures for evaluating when TLD and DRD discrepancies occur. This item remains open. See paragraph 4.f(1) for details.

3. Open Items Identified During This Inspection

(Open) Open Item (298/8232-01): Radiation Worker Training - The licensee has not included all elements of Regulatory Guides 8.27 and 8.29 recommendations into the radiation worker training program. See paragraph 4.c for details.

(Open) Open Item (298/8232-02): Beta Radiation Calibration of Portable Survey Instrumentation - The licensee did not calibrate portable survey instrumentation used to measure beta radiation levels as recommended by ANSI-N323-1978. See paragraph 4.e for details.

(Open) Open Item (298/8232-03): Whole Body Counter Operational Check - The licensee does not perform a whole body counter (WBC) operational check as recommended by ANSI-N343-1978. See paragraph 4.f for details.

(Open) Open Item (298/8232-04): Whole Body Counter Calibration - The licensee had not developed a comprehensive calibration and testing program that satisfies the recommendations of ANSI-N343-1978. See paragraph 4.f for details.

(Open) Open Item (298/8232-05): High Range Containment Monitor Calibration - The licensee does not have an approved electronic calibration procedure for the containment high range radiation monitors. See paragraph 6.c.(3) for details.

(Open) Open Item (298/8232-06): Toxic Gas Release Emergency Procedure - The licensee could not locate an emergency procedure for toxic gas releases. See paragraph 6.e.(3) for details.

(Open) Open Item (298/8232-07): Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 1, "Noble Gas Effluent Monitor" - The licensee has not completed this item. See paragraph 7 for details.

(Open) Open Item (298/8232-08): Item II.F.1, "Additional Accident Monitoring Instrumentation," Attachment 2, "Sampling and Analysis of Plant Effluents" - The licensee has not completed this item. See paragraph 7 for details.

(Open) Open Item (298/8232-09): Item II.B.3, "Attachment No. 1 to Postaccident Sampling System NUREG-0737, Item II.B.3 - Evaluation Criteria Guidelines" - It has not been determined if dissolved oxygen, chloride, and boron analyses satisfy recent evaluation guidelines. See paragraph 6.b for details.

4. Radiation Protection - Operations

a. Qualifications of Personnel

The CNS radiation protection organizational structure is depicted by Figure 1. The licensee maintains 24-hour coverage with health physics and chemistry technicians. All chemistry technicians have been trained in health physics duties and health physics technicians have received training on specific chemistry functions they are required to perform during backshift operations. All technicians, except one hired May 1982, met the recommendations of ANSI 18.1-1971.

The NRC inspectors expressed concern about the limited number of health physics technicians available to support routine operations, maintain 24-hour coverage, and the use of a non ANSI-18.1-1971 qualified individual for backshift health physics coverage. A licensee's representative stated that supervisory personnel who were ANSI-qualified are on call during the backshift and weekends. Technicians have been instructed to contact these persons when off-normal conditions develop.

No violations or deviations were identified.

b. Radiation Protection Audits

The NRC inspectors reviewed the inhouse quality assurance audit performed during the period September 14-26, 1981, in accordance with Procedure QAP-900, "Quality Assurance Plan Chemistry, Health Physics and Environmental Monitoring." All deficiencies and comments (observations) were corrected in a timely manner. The quality assurance department performed a follow-up audit on December 15, 1981, to verify that corrective actions were completed.

The quality assurance department schedules 24 audits each year of various station operations, 4 of these audits are assigned to the

corporate quality assurance department depicted in Figure 2. The corporate quality assurance department had been assigned the responsibility for the annual health physics audit during 1982. This audit was in progress during this inspection.

The NRC inspectors reviewed the two latest safety review and audit board (SRAB) audits performed on April 22, 1980, and August 25, 1981, in accordance with CNS Technical Specification 6.2. These audits were conducted by the director of environmental affairs. The inspectors were not able to determine the health physics technical expertise of this individual. This will be reviewed at a future inspection.

No violations or deviations were identified.

c. Training

The NRC inspectors reviewed the radiation worker training program given to employees, supplemental work force personnel, and contractor-supplied personnel against the requirements of 10 CFR Part 19.12, "Instruction to Workers," and the recommendations of Regulatory Guides 8.13, "Instructions Concerning Prenatal Radiation Exposure"; 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plant"; and 8.29, "Instructions Concerning Risks From Occupational Radiation Exposure."

Training for radiation workers is conducted by the health physics department in accordance with the provisions of Station Procedures 1.5, "Selection and Training of Station Personnel," Revision 8, May 13, 1982, and 9.1.1.1, "Radiation Protection at Cooper Station," Revision 4, May 28, 1981.

The licensee uses a vendor-supplied video tape supplemented with an instructor for plant specific training.

The NRC inspectors noted that all elements of Regulatory Guides 8.27 and 8.29 were not included in the licensee's training program: specifically, bioassays, whole body counting, urinalysis, fecal analysis, avoiding sample contamination, and a practical factors segment. This is considered an open item (298/8232-01) and will be reviewed during a future inspection.

The NRC inspectors verified the training and retraining records of 20 station employees and contractor-supplied personnel were complete.

No violations or deviations were identified.

d. Radiological Protection Procedures

The NRC inspectors reviewed the following procedures which had been revised since the last inspection to determine compliance with 10 CFR Part 20 requirements and recommendations contained in Regulatory

Guides 1.33, 8.8, 8.9, 8.15, and 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.

9.1.1.4 "Special Work Permit," Revision 7, October 6, 1982

9.1.1.5 "Radiography," Revision 4, October 4, 1982

9.1.2.1 "Radiation, Contamination, and Airborne Radioactivity Limits," Revision 9, September 11, 1982

9.1.2.2 "Area Posting and Access Control," Revision 5, June 17, 1982

9.2.1 "Radiation and Contamination Survey Frequency," Revision 8, August 2, 1982

9.2.2 "Radiation Surveys," Revision 10, October 20, 1982

9.3.6.1 "Low Volume and High Volume Air Sampler Operation and Calibration," Revision 7, August 2, 1982

9.1.5 "Respiratory Program," Revision 12, March 12, 1982

Technical Specification 6.3.4 requires procedures to be maintained and consistent with the requirements of 10 CFR Part 20.

When respiratory protective equipment is used to limit the inhalation of airborne radioactive material, a licensee is required to maintain and implement certain basic requirements specified in 10 CFR Part 20.103. One of these requirements specified in 20.103(c)(2) states, in part, " . . . and determination by a physician prior to initial use of respirators, and at least every 12 months thereafter, that the individual user is physically able to use the respiratory protective equipment."

Station Procedure 9.1.5, "Respiratory Protection," Revision 12, Section IV.C.7, states: "The initial medical status of each respiratory user is to be determined by a physician using the Medical Approval Form (Attachment "C"), and a review of the medical status will be performed annually by health physics and documented on Data Sheet (HP-26) located in the health physics files. The review will consist of a discussion with the employee and/or a review of the employee's respiratory records."

Failure to perform a 12-month determination that an individual user is physically able to use the respiratory protective equipment by a physician is considered a violation of 10 CFR Part 20.103(c)(2) (298/8232-02).

e. Instruments and Equipment

The NRC inspectors reviewed the licensee's procedures, calibration, and operation of radiation protection instrumentation against the requirements of the CNS Technical Specifications and recommendations of Regulatory Guides 8.4 and 8.25 and ANSI N13.1-1969 and N323-1978.

The NRC inspectors expressed concern that instruments used to evaluate beta radiation levels were not being calibrated as recommended in Section 4.2 of ANSI N323-1978. This is considered an open item (298/8232-02).

No violations or deviations were identified.

f. Exposure Control

The NRC inspectors reviewed the licensee's exposure control program for compliance with the requirements of 10 CFR Parts 20.101, 20.102, 20.103, 20.104, 20.202, and 20.401; the recommendations of Regulatory Guides 8.4, 8.9, and 8.14; and Industry Standards ANSI 343-1978 and (Draft) N324-1978.

(1) External Exposure Control

The licensee uses a vendor-supplied TLD system and direct reading pocket dosimeters (DRD) to evaluate external radiation exposures. The DRD's are used to evaluate the exposure from gamma radiation on a daily basis and the TLD's for whole body dose measurements on a monthly basis. The licensee has initiated a new method for identifying discrepancies between TLD versus DRD results using the health physics computer. Once a month, the TLD vendor reported exposures are entered into the health physics computer system used to maintain individual exposure histories. The computer will flag discrepancies when the TLD results are 50 percent lower than DRD results or the TLD results exceed DRD results. The licensee has not finalized procedures for determining evaluations or how evaluations will be recorded. This item was identified as an open item (298/8220-01) in NRC Inspection Report 50-298/82-20.

The NRC inspectors verified the licensee was routinely performing a quality control check on vendor-supplied TLD's by exposing 10 TLD's to a cesium-137 source at least quarterly and comparing the reported exposure versus the actual exposure.

(2) Internal Exposure Control

The NRC inspectors reviewed the licensee's WBC procedures (Health Physics Procedures 9.1.8, "Bio-Assay Whole Body Counting")

against the recommendations of ANSI N343-1978, "American National Standard for Internal Dosimetry for Mixed Fission and Activation Products."

The NRC inspectors discussed with licensee representatives the operational check for the WBC system. The licensee does not perform a daily, or when the instrument is in use, check for operability of the WBC system. Procedure 9.1.8 states in Section VI.C.1, "A Functional Check May Be Performed to Verify the Operability of the Whole Body Counter." The ANSI standard in Section 15.3.3(3) recommends that checks shall be performed at least daily while the system is in use and should be made at approximately 8-hour intervals. These checks shall be sufficiently detailed to demonstrate that the measurement equivalent is still in proper calibration and that all electronic components are functioning. This is considered an open item (298/8232-03) pending an operational check of the WBC during usage.

The NRC inspectors reviewed the calibration of the WBC. The licensee performs a "control check" each quarter using a masonite phantom and two radioisotope sources, one cobalt-60, and one cesium-137, of known activity level as described in Procedures 9.1.8, Section VI.C.2. The ANSI standard in Section 15.2 recommends a series of measurements on various standard phantoms loaded with known quantities of radioactivity. These measurements shall be made for the range of organ burdens of interest - for example, 60-20,000 nanocuries of cobalt-60. The NRC inspectors discussed with licensee representatives the need for calibrations to be made at more than one radioactivity level. This is considered an open item (298/8232-04) pending a revision to the current method of calibration.

The licensee was not conducting any respiratory protection training at the time of this inspection. The NRC inspectors reviewed the records of nine station personnel to determine that training and retraining were being performed as required by Health Physics Procedure 9.1.5.

No violations or deviations were identified.

g. Posting, Labeling, and Control

The NRC inspectors, during tours of the licensee's facilities, determined that the licensee was in compliance with the requirements of 10 CFR Parts 20.203b, 20.203e, 20.203f, and 20.207. Station procedures for the posting, labeling, and control of radioactive material, radiation areas, and high radiation areas were adequate. No airborne radioactivity areas were noted.

Radiation work permits were reviewed against licensee surveys and independent measurements made by the NRC inspectors to determine if they afforded an adequate level of protection to workers.

No violations or deviations were identified.

h. Surveys

The NRC inspectors reviewed licensee radiation, contamination, and airborne radioactivity surveys for the period April 14, 1982, through November 28, 1982, to determine compliance with 10 CFR Parts 20.103, 20.201, and 20.401.

The licensee performs routine radiological surveys on a predetermined frequency (daily, weekly, monthly, or quarterly) dependent on the type of area, its use, and radiological conditions. The areas and frequency are detailed in Health Physics Procedure 9.2.1, "Radiation and Contamination Survey Frequency." Special surveys are performed as required for SWP's to evaluate and determine safe working conditions for personnel in specific jobs. The licensee does not routinely make daily surveys of SWP areas unless radiological conditions are uncertain.

The licensee had revised radiation survey procedures and established guidelines for conducting beta radiation surveys on a routine basis. The results of beta surveys were documented on survey forms as required by Health Physics Procedure 9.2.2.

No violations or deviations were identified.

i. Notification and Reports

The NRC inspectors reviewed the records of 20 individuals who terminated employment during November 1982. These individuals had received WBC's as required by Health Physics Procedure 9.1.8, "Bio-Assay Whole Body Counting." The NRC inspectors also reviewed the records of 74 individuals who had terminated since May 1982 to determine compliance with 10 CFR Parts 19.13, 20.407, 20.408, and 20.409. The NRC inspectors' review did not identify any errors or omissions involving termination reports or other reports to any individual.

No violations or deviations were identified.

5. Transportation Activities

The NRC inspectors reviewed the licensee's transportation activities to determine compliance with 10 CFR Parts 20 and 71; 49 CFR Parts 0 through 199; and the recommendations of Regulatory Guides 7.3 and 8.27.

a. Documents Reviewed

- Quality Assurance Plan (QAP)-1200, CNS Quality Assurance Program for Operation, "Quality Assurance Plan - Radioactive Waste Treatment and Disposal," Revision 5, October 3, 1980
- Audit 81-9 conducted March 11-19, 1981
- Audit 82-15 conducted May-July 8, 1982
- Procedure 2.5.4.1, "Solid Wet Waste Packaging, Storage, and Transfer System," Revision 8, October 16, 1981
- Procedure 2.5.4.2, "Solid Wet Waste Drum Filling," Revision 3, October 29, 1980
- Procedure 2.5.4.3, "Radwaste Drum Mixer Operation," Revision 2, December 7, 1978
- Procedure 7.9.2, "Filling Radwaste Containers with Waste," Revision 4, May 11, 1982
- Procedure 9.5.1, "Receival of Radioactive Material," Revision 3, March 25, 1981
- Procedure 9.5.3, "Radioactive Material Shipment," Revision 6, August 6, 1982
- Procedure 9.5.4.2, "Solid Radioactive Shipment," Revision 3, March 6, 1981
- License WN-1019-2, U.S. Ecology, Inc., Expiration Date November 30, 1985, and Attachment, "Listing of U.S. Ecology Clients"
- License 3-11-0043-02, Nuclear Engineering Company, Inc., Expiration Date June 30, 1980, (Timely Renewal Purportedly Pending)

b. Quality Assurance (QA) Audits of Transportation Activities

The licensee's onsite QA audits (81-9 and 82-15) of March 11-19, 1981, and May 1982 through July 8, 1982, respectively, appeared to have been well planned and performed. These audits were quite extensive and corrective actions to identified deficiencies were expediently pursued by station management. The NRC inspectors noted that these audits appear to satisfy their intended purpose to cover the collection, storage, processing, sampling, release, preparation for shipment, and disposal of solid, liquid, and gaseous radioactive

waste materials from the licensee's nuclear facility in accordance with the requirements of station procedures in the area of transportation activities and to verify compliance with NRC and DOT regulations.

No violations or deviations were identified.

c. Management Controls of Transportation Activities

The NRC inspectors noted that the licensee had designated in writing that the health physics supervisor has overall responsibility for receipt, transfer, packaging, and transport of radioactive material; the radioactive waste operator of the station operation staff is responsible for the solidification and processing of radioactive waste; the senior site mechanical engineer of the station maintenance staff is responsible for the packaging of compactible and noncompactible radioactive waste; the technicians of the station health physics staff are responsible for the loading and transport aspects of radioactive waste; and the health physicist is responsible for the coordination of the transportation activities.

No violations or deviations were identified.

d. Radioactive Waste Training

The NRC inspectors reviewed the licensee's program for indoctrination and training of personnel performing transport activities. The NRC inspectors determined that the licensee's program was adequate to assure that proficiency is achieved and maintained.

The NRC inspectors noted that the licensee's letter dated August 31, 1979, in response to IE Bulletin 79-19 stated, in part, in Items 5 and 6 that training records will be available for future inspection by NRC personnel. The NRC inspectors also noted that training records were not available during this inspection for retraining of two individuals involved in the transfer, packaging, and transport of radioactive wastes.

This is considered a deviation. (298/8232-01)

No violations were identified.

e. Selection of Packagings

The NRC inspectors reviewed records of the licensee's determination that packagings had been fabricated in accordance with the approved design and noted that the licensee's packaging design was as specified in an NRC certificate or a DOT specification.

The NRC inspectors examined several fabricated packagings and noted conformance with the required DOT specification markings or certificate markings.

The NRC inspectors noted that the licensee had not made any Type B or fissile radioactive materials shipments during the period covered by this inspection.

No violations or deviations were identified.

f. Preparation of Packages for Shipment

The NRC inspectors noted that the licensee, in accordance with written procedures, assures that prior to each use the package: is proper for the contents to be shipped; is in unimpaired physical condition; is properly marked (i.e., "RADIOACTIVE LSA" if of low specific activity radioactive materials shipped in exclusive-use vehicle); complies with the radiation level limits at the surface of the package, at 3 feet (transport index); and is such that the removable non-fixed contamination is below the regulatory limits.

No violations or deviations were identified.

g. Shipments

The NRC inspectors observed the loading of packages on three transport vehicles (exclusive-use). The packages were labeled "RADIOACTIVE LSA" and the transport vehicle was placarded "RADIOACTIVE." Shipment 82-72 was shipped December 1, 1982, whereas Shipments 82-73 and 82-74 were shipped December 2, 1982. The shipments were also being observed and independently measured by a third party representative of the Nevada Inspection Services as a condition of acceptance by the Nuclear Engineering Company, Inc. The NRC inspectors verified that the radiation levels around the transport vehicles were within the required limits and the shipping papers included the following items of information:

- DOT proper shipping name
- Class of the hazardous material
- The name of each radionuclide
- A description of the physical and chemical form
- The activity contained in each package
- The category of label applied and the transport index assigned to each package

. Instructions for maintenance of exclusive-use shipment control

. Shippers' certification

No violations or deviations were identified.

h. Records

The NRC inspectors reviewed the licensee's radioactive material receiving and shipping records. The receipt inspection of radioactive materials is the responsibility of the health physics staff and appeared to satisfy the requirements of 10 CFR 20.205. The offsite shipment records appeared to meet NRC and DOT regulations as to required information.

No violations or deviations were identified.

i. Previous Radioactive Waste Shipments

10 CFR Part 71.5(a), "Transportation of Licensed Material," requires that no licensed material shall be transported outside of the confines of his plant unless the requirements of the regulations appropriate to the mode of transportation of the Department of Transportation in 49 CFR Parts 170 through 189 are met. Section 173.393, "General Packaging and Shipment Requirements," states in paragraph (j)(3) that the radiation dose rate should not exceed 10 millirem per hour at any point 2 meters (6 feet) from the vertical planes projected by the outer lateral surface of the vehicle.

On July 7, 1982, the licensee was notified by the State of Nevada, Department of Human Resources, Division of Health, Bureau of Consumer Health Protection Services, that on June 25, 1982, a shipment of radioactive waste from CNS was received at the Beatty, Nevada, site and found to have radiation levels at 6 feet from the side of the trailer in excess of 10 millirem per hour. This situation constituted a violation of Department of Transportation Regulation 49 CFR Part 173.393(j) and Article 2.5.2.1 of the State of Nevada Regulations governing use of state-owned area for disposal of radioactive waste.

The NRC inspectors informed licensee representatives that failure to meet requirements of Department of Transportation Regulations constitutes a violation of 10 CFR Part 71.5(a). (298/8232-01)

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

(c) Letter, November 20, 1979, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(d) Letter, January 11, 1980, to H. R. Denton (USNRC) from J. M. Pilant (NPPD)

(e) Letter, April 10, 1980, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(f) Letter, September 5, 1980, to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits from D. G. Eisenhut (USNRC)

(g) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(h) Letter, July 10, 1981, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

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

(j) Letter, April 16, 1982, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

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

(l) Letter, April 16, 1982, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(m) Letter, June 4, 1982, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(n) Administrative Procedure 1.13, Attachment E, Minor Design Change, MDC 80-086

(o) Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974

(p) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978

(q) Standard Review Plan, Section 15.6.5, "Loss-of-Coolant Accidents (LOCA) Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary"

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

(s) 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 1.3 (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 Emergency Core Cooling System (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 TSC are areas that will require full-time occupancy ($\leq$15 mrem/hr averaged over 30 days) during the course of the accident.

(3) Conclusions

The licensee performed a radiation and shielding design review. This "inhouse" study used the source term per Regulatory Guides 1.3 and 1.7, Standard Review Plan 15.6.5, but the methodology made the results conservative. This review considered the following areas:

- Radwaste Building
- Reactor Waste Sample Station
- Radwaste Building Torus Vent
- Torus Vent
- Torus Vent Line
- Drywell Vent, Elevation 966-998 feet
- Drywell Vent (horizontal)
- Radwaste Contol Room
- Emergency Control Center
- Radiochemistry Laboratory
- Control Room
- Technical Support Center
- Turbine Building

The continuous occupied control room exposure rate was calculated to be 12.3 mr/h when an accident occurs and 2.6 mr/h after 12 hours. Also, the exposure rate in the TSC during an accident situation is less than 13.7 mr/h. These exposure rates meet the guidelines of the General Design Criteria 19 where a vital area for continuous occupancy (averaged was 30 days) must not have a dose equivalent rate to exceed 15 mrem/h. CNS is a boiling water reactor (BWR) and the reactor building will be evacuated during a LOCA and no personnel will be allowed to enter.

The only modification resulting from the shielding design review study is the installation of the postaccident sampling system (PASS) lines (see paragraph 6.b) from the sampling stations to

outside the reactor building into the radwaste building. This modification was needed because of the projected excessive dose rates received during sample collection in an accident situation in the reactor building. The modification was performed according to the minor design change, MDC 80-086.

The review responsibility of radiation qualification for the 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)(s)).

It was concluded during the inspection that the shielding design review portion of this item (Item II.B.2) for CNS met the NUREG-0737 criteria and is, therefore, considered acceptable and closed.

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

(1) Documents Reviewed

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

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

(c) Letter, November 20, 1979, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

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

(e) Letter, January 11, 1980, to H. R. Denton (USNRC) from J. M. Pilant (NPPD)

(f) Letter, April 10, 1980, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(g) Letter, September 5, 1980, to All Licensees of Operating Plants and Applicants for Operating Licensees and Holders of Construction Permits from D. G. Eisenhut (USNRC)

(h) Memorandum, October 24, 1980, to Region Directors (USNRC) from S. E. Bryan (USNRC)

(i) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(j) Letter, February 27, 1981, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(k) Letter, July 10, 1981, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(l) Memorandum, December 21, 1981, to T. M. Novak (USNRC) from W. E. Kreger (USNRC)

(m) Letter, December 28, 1981, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

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

(o) Letter, April 16, 1982, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

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

(q) Letter, August 9, 1982, to J. M. Pilant (NPPD) from D. B. Vassallo (USNRC)

(r) Letter, September 1, 1982, to D. B. Vassallo (USNRC) from J. M. Pilant (NPPD)

(s) Letter, August 26, 1981, to L. C. Lessor (NPPD) from W. C. Jones (OPPD)

(t) Flow Diagram, "Radwaste Building Heating and Ventilating," Drawing 2021

(u) Flow Diagram, "Radio Chemistry Laboratory Heating Ventilation Air Conditioning," Drawing 2024, Sheet 2

(v) Radio Chemistry Procedure 8.4.1.1, "PASS Reactor Coolant and Containment Atmosphere Sampling"

(2) Discussion

NUREG-0737 states:

"A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 or 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

"A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify certain radionuclides that are indicators of the degree of core damage. (The combined time must be 3 hours or less for sampling and analysis.) Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

"In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift)."

NUREG-0737 clarifies the above by stating the following:

"The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

"The licensee shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame, quantification of the following:

- "certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases; iodines and cesiums, and nonvolatile isotopes);

- "hydrogen levels in the containment atmosphere;

- "dissolved gases (e.g., H_2), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids.

- "alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

"Reactor coolant and containment atmosphere sampling during postaccident conditions shall not require an isolated auxiliary system [e.g., the letdown system, reactor water cleanup system (RWCUS)] to be placed in operation in order to use the sampling system.

"The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

"The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities).

"The analysis of primary coolant samples for boron is required for PWR's. (Note that Revision 2 of the Regulatory Guide 1.97, when issued, will likely specify the need for primary coolant boron analysis capability at BWR plants.)

"Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

"The licensee's radiological and chemical sample analysis capability shall include provisions to:

. "Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guides 1.3 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 uCi/g to 10 Ci/g.

. "Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around

samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

"Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

"In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

. "Provisions of purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the reactor coolant system or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

. "The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters."

(3) Conclusions

The NRC inspectors determined from the licensee's past experience that a reactor coolant and containment atmosphere sample could be collected and analyzed within 3 hours from PASS. The licensee has the ability to analyze the samples for the necessary radioisotopes to determine the extent of core damage that may occur during an accident.

From the radiation and shielding design review performed in 6.a, the exposure rate in the radiochemistry laboratory could be as great as 14.5 mr/h when accident occurs and reduces to 3.0 mr/h in 12 hours. Therefore, individuals performing sampling and analytical functions during an accident would not receive a dose equivalent in excess of 3 and 18.75 rem to the whole body or extremities, respectively. Also, the dose rate in the radiochemistry laboratory is not prohibitive to making an accurate analysis.

The licensee is able to perform chemical analyses of the reactor coolant for chloride and boron in the radiochemistry laboratory within the NUREG-0737 prescribed time limits. There are dedicated hydrogen and oxygen monitors in the drywell which read-out in the control room in percent of hydrogen and oxygen.

It was determined that no auxiliary system is required to be isolated when any sample is collected from the PASS.

The licensee has a written agreement (6.b.(1)(s)) with the Omaha Public Power District where Fort Calhoun Nuclear Station's facilities will be available for analyzing samples in case they are needed.

The PASS has 0.25 and 0.5 inch O.D. stainless steel tubes for liquid and gas, respectively, running from the plant's sampling station in the reactor building to the reactor building side of the wall separating the reactor building and the radwaste building. On the reactor building side of the wall are installed a dilution tank, valves, tubing, and return lines. The sampling and return lines run through the wall to the PASS sampling station in the radwaste building. The PASS provides for remote sampling of the reactor coolant water, torus water, and containment atmosphere under conditions where reactor building entrance is prohibited. The reactor coolant water sample, from the recirculation loop, and the containment atmosphere sample may be flushed for a representative sample, diluted, transferred to a collection vessel outside the reactor building, removed to the radiochemistry laboratory, and tested in a period of less than 3 hours. Conductivity and pH analysis of undiluted samples are provided by inline samplers. Total gas is determined by expanding a representative sample into a known volume. Additionally, the ability to sample, dilute, and analyze torus water (through the RHR system) and reactor water (from the reactor water cleanup system) is provided. Any of the above samples may also be collected undiluted. Remote sample system operation is performed from a panel in the radwaste building. Sample system wastes are collected and subsequently transferred to the primary containment.

The samples to be analyzed in the radiochemistry laboratory are always diluted to where the exposure rates are approximately 10 mr/h which restricts the background radiation in the radiochemistry laboratory.

The licensee maintains the dilution method meets the intent of NUREG-0737 for the measurement of nuclide concentration over the range of 1 uCi/g to 10 Ci/g, such that a sample with too high specific activity for the instrumentation to adequately analyze can be diluted to the necessary level compatible with the instrumentation.

Also, dilution is used to such an extent that the licensee maintains, because of small amounts of radioactivity, the ventilation exhaust is adequately filtered with HEPA filters.

The radiochemistry analytical instrumentation has the necessary capability (accuracy, range, sensibility, and radioisotope library) to promptly qualify the radionuclides that are indicators of the degree of core damange that might occur during an accident.

The NRC inspectors noted that there may be some effects of the long length of PASS lines from the systems that are being sampled to the PASS sampling station. The lines are 462, 397, 522, and 304 feet to the reactor water recirculation system, reactor water cleanup system, residual heat removal system, and containment atmosphere, respectively. After each sample is taken the line is flushed or purged with at least one volume of that specific sample system. The licensee has sampled these systems at the reactor building sampling station and at the PASS sampling station during normal operating conditions and the results were the same.

Under accident conditions with more radioactivity in the lines, there was concern by the NRC inspectors that there could be problems with plateout, blockage of sample lines and sample distortion in the long 0.25-inch outside diameter (O.D.) liquid line (the 0.5-inch O.D. gas line is "heat traced" to prevent plateout of the iodines). Upon conversing with Messrs. Byron Siegel and James Wing of the Office of NRR on December 21, 1982, these concerns were alleviated.

Based on the criteria appearing in the original NUREG-0737, the licensee appears to satisfy the requirements. However, it was not determined if additional analytical procedures need to be developed to meet recent evaluation criteria guidelines. See reference 6.b.(1)(q). This item is considered open pending further review by NRR and the Regional Office. (298/8232-09)

No violations or deviations were identified.

c. Item II.F.1 Attachment 3, "Containment High-Range Radiation Monitor"

(1) Documents Reviewed

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

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

(c) Letter, November 20, 1979, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(d) Letter, April 10, 1980, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(e) Letter, June 30, 1981, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(f) Memorandum, December 21, 1981, to T. M. Novak (USNRC) from W. E. Kreger (USNRC)

(g) Letter, December 29, 1981, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(h) Stone and Webster Engineering Corporation, Sketch 1309.12-55(d)-1 (901' 4¼" elevation drywell floor plan)

(i) Victoreen High Range Containment Monitor, Drawing 877-1, "Qualification Summary," 950.301

(j) CNS Health Physics Procedure 9.4.4, "High Range Containment Monitor, Victoreen Model 875, Calibration"

(k) Letter, April 30, 1981, to J. L. Scheer (NPPD) from K. E. Stafford (Victoreen), "Containment Monitor Qualification Test Plan"

(l) Victoreen, Inc., "High Range Containment Area Monitor Detector Energy Response Curve," Drawing 877-1

(m) Letter, March 31, 1981, to L. Lessor (NPPD) from K. E. Stafford (Victoreen) (Calibration and Qualification Testing)

(2) 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 and accident.

The specification of 1E+08 rad/hr in the above position was based on a calculation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments, but gamma-sensitive instruments can be so qualified. 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 1E+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.

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

(3) Conclusions

The licensee has installed a Victoreen Containment High Range Area Radiation Monitor System, Model 875. There are two detectors, Model 877-1, located in the drywell at the 901 feet 9.25 inch elevation 180° apart. The detectors are read out in the control room by Victoreen Model 876A-1 (RMA-RM-40 A & B) instruments and recorded on a strip chart, RMA-RR-40.

This system has a range from 1 to 1E+07 R/h for gamma radiation, therefore, the exposure rate can be followed through accident and postaccident situations.

The NRC inspectors reviewed the design and qualification criteria of the detectors for functioning in an accident environment. The results of the qualification tests indicated these monitors would function properly during an accident and in postaccident conditions.

The Victoreen Containment High Range detectors have the ability to detect 60 keV gamma radiation with a linear energy response that meets NUREG-0737 requirements of ± 20 percent for photons of 0.1 to 3 MeV.

These monitors were calibrated by the vendor, Victoreen Instrument Corp., before delivery. They were calibrated at 1950 R/hr, 210 R/hr, 76 R/hr, and 1900 R/hr, 200 R/hr, 90 R/hr for channels A and B, respectively, with radiation sources. They were also calibrated electronically at 1 R/hr, 800 R/hr, E+03 R/hr, and E+07 R/hr.

The licensee performed a preoperational calibration on the high range containment monitors with a calibrated source at 3.5 R/hr, 7.5 R/hr, 130 R/hr, and 3.7 R/hr, 7.8 R/hr, and 116 R/hr on channels A and B, respectively. These calibrations were performed according to Health Physics Procedures 9.4.4 after the monitors were installed. The licensee also calibrated these monitors electronically at E+01, E+02, E+03, E+04, E+05, E+06, and E+07 R/hr. The licensee performed the electronic calibration per the vendor's calibration procedure in lieu of a station-approved calibration procedure. Therefore, this is an open item (298/8232-05) pending a station approved electronic calibration procedure.

These monitors are scheduled to be recalibrated during each refueling outage. They have been calibrated and recalibrated in October 1981 and June 1982, respectively.

Three instrumentation and control technicians have received 4-16 hours training from a Victoreen representative on the electronic calibration of these high range containment monitors. The health physics calibration training consists of technician rotation at each time of calibration.

No violations or deviations were identified.

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

(1) Documents Reviewed

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

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

(c) Letter, November 20, 1979, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(d) Letter, January 11, 1980, to H. R. Denton (USNRC) from J. M. Pilant (NPPD)

(e) Letter, April 10, 1980, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(f) Letter, September 5, 1980, to All Licensees of Operating Plants and Applicants for Operating Licensees and Holders of Construction Permits from D. G. Eisenhut (USNRC)

(g) Memorandum, October 24, 1980, to Region Directors from S. E. Bryan (USNRC)

(h) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(i) Letter, July 10, 1981, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(j) Letter, February 22, 1982, to J. M. Pilant (NPPD) from D. B. Vassallo (USNRC)

(k) Chemistry Procedure 8.2.1, "Chemical Analysis and Instrument Calibration Schedule"

(l) Chemistry Procedure 8.5.2.5, "Gamma Spectrometer (Operation-Calibration Procedure)

(n) Chemistry Procedure 8.4.1.2, "Emergency Sampling Gaseous Release"

(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 for the 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 licensee's Tracon Northern - 11 GeLi System located in the radiochemistry laboratory.

This system is used to analyze the iodine samples. According to Chemistry Procedure 8.2.1, this system is calibrated daily or prior to use with Ba-133, Cs-137, and Co-60 sources. The NRC inspectors noted the Ba-133 gamma of 356 keV is used for calibration instead of the 365 keV gamma radiation of I-131 as recommended in NUREG-0737.

The licensee utilizes a Gelman glass fiber filter, Type A, preceeding the silver zeolite cartridge (Model "C", 5A351B, F&J Speciality Products) or charcoal cartridge (CESCO 8170, SC 727) for iodine collection in a Radeco H-809VI air sampler. The licensee has available for iodine collection four Radeco H-809VI and two Bendix air samplers. The license has the necessary capability to flush the charcoal cartridge of any entrapped noble gases with nitrogen gas.

Since the emergency iodine sampling and analysis are performed using the normal procedures and equipment, the training for normal iodine sampling and analysis is adequate.

It is concluded that this item (Item III.D.3.3) meets the NUREG-0737 criteria and is considered 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 30, 1979, to All Operating Nuclear Power Plants from H. R. Denton (USNRC)

(c) Letter, November 20, 1979, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(d) Letter, January 11, 1980, to H. R. Denton (USNRC) from J. M. Pilant (NPPD)

(e) Letter, April 10, 1980, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(f) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(g) Letter, July 10, 1981, to J. M. Pilant (NPPD) from T. A. Ippolito (USNRC)

(h) Letter, February 24, 1982, to J. M. Pilant (NPPD) from D. B. Vassallo (USNRC)

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

(j) Letter, June 4, 1982, to D. G. Eisenhut (USNRC) from J. M. Pilant (NPPD)

(k) 10 CFR Part 50, Appendix A (General Design Criteria for Nuclear Power Plants), Criterion 19, "Control Room"

(l) Standard Review Plan 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity"

(m) Standard Review Plant 2.2.3, "Evaluation of Potential Accidents"

(n) Standard Review Plan 6.4, "Habitability Systems"

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

(p) Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"

(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. The new clarification specifies that licensees that meet the criteria of the SRP's 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 SRP:

2.2.1-2.2.2	"Identification of Potential Hazards in Site Vicinity";
2.2.3	"Evaluation of Potential Accidents"; and
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 LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and Appendix 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) Conclusions

Enclosure 5 of the December 30, 1980, letter (see 6.e.(1)(f)) is the licensee's control room habitability study. This study included the following:

- Site Characteristics
  Geography, Plant Layout and Control Room Characteristics
  Design Basis Methodology
  Type and Location of Potential Toxic Gas Hazard

- Design Basis Radiology
  Methodology
  Results

- Toxic Gas Review
  Methodology
  Results
    Concentration Plots for:
    Chlorine Truck Accident
    Chlorine Train Accident
    Anhydrous Ammonia Barge Accident
    Carbon Dioxide Accident
    Nitrogen Accident
    Sulfuric Acid Accident
    Sodium Hydroxide Accident
    Ammonia Barge Accident

- Control Room Protection
  Ventilation Systems
  Emergency Provisions

This control room habitability study addresses the requirements of NUREG-0737 and finds the existing CNS control room envelope and habitability systems to be adequate. In this study, the systems have been evaluated, analyzed, and determined adequate to protect the control room operators against the effects of an accidental release of either toxic or radioactive gas thereby allowing the nuclear power plant to be safely operated or shut down under design basis accident conditions.

Section 6 of this study states, "An emergency procedure will be written by January 1, 1982, which will discuss the necessary actions and responsibilities for toxic gas releases in the plant vicinity." The licensee was unable to produce this procedure and this will be considered an open item (298/8232-06) pending the review of the procedure.

No violations or deviations were identified.

7. NUREG-0737 Items Not Completed

The NRC inspectors were unable to review the following items: 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.

These items (298/8232-07 and 298/8232-08) are considered open pending completion of 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 Regulations of the NRC informing them of delays.

8. Exit Interview

The NRC inspectors met with the license representatives identified in paragraph 1 at the conclusion of the inspection on December 3, 1982. The NRC inspectors discussed the scope and findings of the inspection.