U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

Report: 50-267/82-21

License: DPR-34
Docket: 50-267

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

Facility: Ft. St. Vrain Nuclear Generating Station (FSV)

Location: Platteville, Colorado

Inspection Conducted: August 30 - September 3, 1982

Inspectors: R. E. Baer, Radiation Specialist — 12/22/82 Date

for W. L. Holley, Radiation Specialist — 12/22/82 Date

Approved by: Blaine Murray, Chief, Facilities Radiation Protection Section — 12/22/82 Date

Inspection Summary

Inspection conducted August 30 - September 3, 1982 (Report 50-267/82-21)

Areas Inspected: Routine, unannounced inspection by regional based inspectors of transportation activities, radiation protection operation and select NUREG-0737 items including: management controls; preparation of packages for shipment; delivery of completed packages to carrier; receipt of packages; incident reporting; indoctrination and training program; audit program; recordkeeping; radiation protection audits; radiation protection training; radiation protection procedures; exposure control; and posting, labeling and control. The inspection involved 66 onsite hours by two inspectors.

Results: No violations or deviations were identified. Six open items are discussed in paragraphs 5.f, 6.e, 7.b, 7.d, 7.e., and 7.f. One unresolved item is discussed in paragraph 6.c.

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DETAILS

1. Persons Contacted

   a. Public Service Company of Colorado (PSCo)

      *E. D. Hill, Station Manager
      *T. J. Borst, Radiation Protection Manager
      *W. S. Franek, Site Engineering Manager
      C. Fuller, Technical Services Engineering Supervisor
      *J. W. Gahm, Quality Assurance Manager
      R. E. Huster, Quality Assurance Auditing Coordinator
      V. McGaffic, Radiochemical Supervisor
      *L. W. Singleton, Quality Assurance Operations Superintendent
      *T. E. Schleiger, Health Physics Supervisor
      *R. Wadas, Training Supervisor
      W. E. Woodard, Health Physicist

   b. Other Personnel

      *G. L. Plumlee III, NRC Resident Inspector

      The NRC inspectors also interviewed several other licensee employees, including health physics, radiochemistry technicians and administrative personnel.

   *Denotes those present during the exit interview.

2. Licensee Action on Previous Inspection Findings

   (Closed) Open Item (267/8115-01) - Health Physics Technicians Training and Experience: This item was discussed in NRC Health Physics Appraisal Report 50-267/80-13 and NRC Inspection Report 50-267/81-15 and involved the lack of adequate guidance to ensure that contract health physics technicians training and experience meet ANSI N18.1-1971 criteria. The licensee revised Procedure HPP-46, Section 4.7 and defined 2 years to mean 24 months experience in the specialty. This item is considered closed.

   (Closed) Open Item (267/8115-02) - Installation of Personnel Monitoring Equipment: This item was discussed in NRC Health Physics Appraisal Report 50-267/80-13 and NRC Inspection Report 50-267/81-15 and involved the lack of personnel monitoring equipment at the exit from the protected area. The licensee purchased and installed two walk-through portal monitors. One monitor was installed at the exit to the reactor building, the other at the security building. This item is considered closed.

   (Closed) Unresolved Item (267/8115-03) - Contractor Health Physics Qualifications: This item is discussed in NRC Inspection Report 50-267/81-15 and involved several technicians who were given credit toward the 2 years of experience through training programs and overtime work on the job.

The licensee stated the technicians in question have terminated and have been replaced by personnel who meet the 24-month experience criteria in accordance with Procedure HPP-46. This item is considered closed.

(Closed) Violation (267/8128-01) - Titanium Sponge: This item was identified in NRC Inspection Report 50-267/81-28 and involved the titanium sponge in the helium purification system which had been out of service for 18 months. The NRC inspectors verified by visual inspection that the titanium sponge had been installed and was in service as required by Technical Specification LCO 4.8.1.c. This item is considered closed.

(Open) Unresolved Item (267/8128-01) - Reactor Building Exhaust Filters: This item was identified in NRC Inspection Report 50-267/81-28 and involved the documentation of filter tests required by Technical Specifications. The licensee stated that the reactor building exhaust filters had been tested to meet the requirements of the Technical Specifications during the system start-up testing. This will be reviewed during a future inspection.

3. Open Items Identified During This Inspection

(Open) Open Item (267/8221-01) - 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 6.e for details.

(Open) Open Item (267/8221-02) - Radioactive Waste Retraining Program: The licensee had not developed a formal training/retraining program for personnel involved in the transfer, packaging, and transport of radioactive materials. See paragraph 5.f for details.

(Open) Open Item (267/8221-03) - Primary Coolant Sample Lines: The licensee had not determined potential for line blockage, activity plate-out or sample distortion. See paragraph 7.b for details.

(Open) Open Item (267/8221-04) - Noble Gas Effluent Monitors: The licensee had not determined that the noble gas effluent monitors met ANSI N13.1 design criteria. See paragraph 7.d for details.

(Open) Open Item (267/8221-05) - Reactor Building Ventilation Exhaust Monitor: The license had not determined effect of entrained moisture on iodine sampling capabilities. See paragraph 7.e for details.

(Open) Unresolved (267/8221-06) - Radiation Worker Training Program: The licensee had not revised the radiation worker training program to include the recommendations of Regulatory Guides 8.27 and 8.29. See paragraph 6.c for details.

(Open) Open Item (267/822-07) - Containment High Radiation Monitors: The licensee had not determined operability of the containment radiation monitors during elevated temperature conditions following an accident. See paragraph 7.f for details.

4. Regulatory Documents

The NRC inspectors verified that the licensee had current copies of applicable Nuclear Regulatory Commission (NRC) and Department of Transportation (DOT) regulations so as to be able to comply with their requirements.

The licensee subscribes to Dat-O-Line, Inc., Charleston, South Carolina, radioactive waste management service which provides current copies of 10 CFR Part 71 (NRC), 40 CFR (DOT), 39 CFR (Postal Service), and State and nonfederal regulations. Additional information is provided about Notices, Pending Rules, and Proposed Rules as prepared by the DOT and other authorities and extracted from the Federal Register. All the above categories are updated with a biweekly supplement.

No violations or deviations were identified.

5. Transportation Activities

a. Management Controls

The management control system for radioactive material management is described in general in Administrative Procedure Q-1 and more specifically in Procedure P-3. The health physics supervisor is designated as the individual with the responsibility to insure the proper shipment and receipt of all radioactive material to and from the plant. The health physics department is responsible for the collection, compaction or solidification, preparation of the shipment and loading of the transport vehicle. The licensee generates a minimal quantity of radioactive waste material from maintenance activities and, therefore, does not provide dedicated personnel for radioactive waste activities.

The licensee had developed and implemented procedures for the various processes and details of the radioactive material handling program. These procedures included:

HPP-23, "Receiving Radioactive Materials," Issue 6

HPP-26, "Radioactive Material Control and Handling," Issue 4

HPP-30, "Radioactive Material Classification, Packaging, and Labeling," Issue 0

The quality assurance - operations department is responsible for planned and periodic audits of the radioactive waste management program. The licensee had developed Procedure Q-18, "Quality Assurance Audit and Monitoring Program," Issue 5, to provide guidance in implementing these audits. Audits are scheduled on a biannual frequency.

No violations or deviations were identified.

b. Preparation of Packages for Shipment

The licensee's program for preparation of by-product radioactive material for shipment was reviewed against the requirements of 10 CFR Parts 71.12, 71.31, 71.35, 71.53 and 71.54, 49 CFR Parts 172 and 173, and the following generally accepted codes, guides and standards:

Regulatory Guide 7.1 - Administrative Guide for Packaging and Transporting Radioactive Material.

Regulatory Guide 7.4 - Leaking Tests on Packages for Shipment of Radioactive Materials.

ANSI N14.10.1 - Administrative Guide for Packaging and Transporting Radioactive Materials.

ANSI N14.5 - Leak Tests on Packages for Shipment of Radioactive Materials.

The licensee had developed and implemented procedures for preparation of radioactive materials for shipment. These procedures (see list in paragraph 5) included requirements for visual inspection prior to filling or loading the package; marking of package weight and contents; labeling requirements appropriate for the type of package; and radiation and contamination limits for packages.

The NRC inspectors noted by observation of the radioactive waste compaction and storage facility that the licensee used, for shipments of low specific activity radioactive waste, steel drums manufactured in accordance with DOT specification 17H (49 CFR Part 178.118).

The licensee had not made a shipment of low specific activity radioactive waste since receiving their operating licensee in 1973.

No violations or deviations were identified.

c. Delivery of Completed Packages to Carrier

The licensee's program for delivery of completed packages to a carrier for transport was reviewed against the requirement of 10 CFR Part 71.55 and 49 CFR Parts 100 to 199. Activities for delivery of completed packages to a carrier were governed by previously mentioned Procedure HPP-30.

The NRC inspectors examined this procedure for consistency with regulatory requirements and to determine whether it covered all aspects. The licensee had not shipped any radioactive waste, therefore, records were not available to verify adherence to procedural requirements.

No violations or deviations were identified.

d. Receipt of Packages

The licensee's program for the receipt of packages containing radioactive material was examined against the requirements of 10 CFR Part 20.205 and conformance to Procedure HPP-23.

The NRC inspectors reviewed this procedure for compatability with regulatory requirements and to determine whether it covered all aspects of the work being carried out.

No violations or deviations were identified.

e. Incident Reporting

The NRC inspectors reviewed the licensee's procedures for incident reporting against the requirements of 49 CFR Parts 171.15 and 171.16. The reporting of incidents were not covered by plant procedures. The licensee had not offered for shipment any radioactive waste material, and plans on using a contract carrier when a shipment is made.

No violations or deviations were identified.

f. Indoctrination and Training Program

The licensee's indoctrination and training program, as it pertains to the packaging of low level radioactive waste for transport and burial, was examined against the provisions of IE Bulletin No. 79-19 and the licensee's response to this bulletin.

The NRC inspectors reviewed documentation of training conducted since January 1980 for personnel involved in the transfer, packaging, and transport of radioactive material.

Two members of the health physics staff had attended a vendor conducted workshop on packaging and transportation of radioactive materials. Two additional staff members are scheduled to attend this workshop in the fall of 1982.

The NRC inspectors noted that health physics personnel receive training in radioactive waste systems and processes and is documented in the individual's "Health Physics Technician Training Check-off List," but the licensee had not developed a formal retraining program for personnnel involved in the transfer, packaging, and transport of radioactive materials. The licensee's station training program administrative manual, HPC-2 states in Section 4.4.3.a, "The Health Physics and Radiochemistry Department Retraining program is conducted as considered necessary by the radiation protection manager and the training supervisor." This item is considered open pending implementation of a suitable training and retraining program which details retraining frequency and subject material to be presented (267/8221-02).

No violations or deviations were identified.

g. Audit Program

The licensee's audit function for the low-level radioactive waste transfer, packaging, and transport activities was examined against the requirements of 10 CFR Part 71 and IE Bulletin No. 79-19 and within the framework of the following generally accepted guidance:

-Regulatory Guide 1.33 - Quality Assurance Program Requirements

-ANSI N18.7-1976 - Administrative Controls and Quality Assurance for Operational Phase of Nuclear Power Plants.

The NRC inspectors reviewed the audits of transportation activities, including the latest audit, conducted by the licensee:

QAA-1501-79-02, dated September 24-26, 1979 QAA-1501-81-01, dated August 19 - September 3, 1981

These audits were conducted in accordance with the licensee's written procedures listed in paragraph 5 and included a checklist for the areas reviewed. Deficiencies identified during these audits, recommendations and comments relating to the areas audited were contained on Form QAA-602. All deficiencies were corrected in a timely manner. The inspectors also reviewed audits QAA-501-80-01, dated June 3 - August 13, 1980, and QAA-501-82-01, dated August 11-31, 1982, which were conducted on spent fuel shipments.

No violations or deviations were identified.

h. Recordkeeping

The licensee had not made a shipment of low-level radioactive waste. No records were available for review to determine compliance with the requirements of 10 CFR Part 71.62.

No violations or deviations were identified.

i. Spent Fuel Shipping Program

1) Responsibility

The responsibility for Special Nuclear Material (SNM) has been delegated to the Technical Services Department in Administrative Procedure G-6, "Control of Special Nuclear Material," Issue 6. Section 4.1.3 states, "Technical Services prepares and controls all transmittal forms necessary for the transfer or possession of Special Nuclear Material." The reactor engineer has been assigned the responsibility for SNM documentation.

2) Procedures

The licensee has developed and implemented procedures for all phases of fuel handling; these are designated Fuel Handling Procedures (FHP). Specifically, Procedures FHP-5 and FHP-6 relate to the shipment of spent fuel and cover the handling, loading, and inspection (including checklists) of the spent fuel cask.

3) Spent Fuel Shipments

The licensee had made 12 shipments during 1982 of spent fuel to the Department of Energy (DOE) contractor-operated facility in Idaho. The licensee had a copy of a letter from the Public Service Company of Colorado, dated April 14, 1982, to the DOE contractor requesting license information to receive spent fuel and the reply letter from the Idaho Operations Office DOE, dated May 6, 1982, which stated the contractor was authorized to receive spent fuel.

4) Spent Fuel Shipping Container

All shipments of spent fuel had been made in shipping containers designed USA/6346/B Model FSV-1. A Certificate of Compliance, Number 6346, Revision 4, dated September 25, 1980, which pertained to these containers was available for review. This Certificate of Compliance expires September 30, 1985.

5) Notifications And Reports

The NRC inspectors reviewed records for the advanced Notification of Governors of states through which spent fuel was being transported and Region IV, as required by 10 CFR Parts 71.5b and 73.72. The licensee had made the proper notification prior to scheduled shipments.

The NRC inspectors reviewed select spent fuel shipment documentation for shipments made during 1982.

No violations or deviations were identified.

6. Radiation Protection Operations

a. Radiation Protection Personnel Staffing and Qualifications

The NRC inspectors reviewed the station organization to determine if there had been any changes affecting the radiation protection program and examined the staffing level of the health physics department. The licensee's organization and staffing level are depicted below:

1 - Radiation Protection Manager (1)*

1 - Health Physics Supervisor (1)

1 - Health Physicist (1)

1 - Senior Health Physics Technicians (1)

7 - Health Physics Technicians (6)

*The numbers in parentheses denote the present staffing level.

The NRC inspectors reviewed the resumes and training records of the three supervisory level and all seven of the licensee's staff health physics technicians. All health physics technicians met the selection and qualification criteria of ANSI-N18.1-1971, and the radiation protection manager and health physics supervisor met the recommendations of Regulatory Guide 1.8. The licensee has supplemented station health physics technicians with four contractor-supplied health physics technicians. The NRC inspectors reviewed the resumes and training records of these personnel. Three of these persons did not meet the qualification criteria, but were not assigned to function in positions having senior health physics responsibilities.

No violations or deviations were identified.

b. Radiation Protection Audits

The NRC inspectors reviewed the licensee's audit program relating to radiation protection operations conducted by the quality assurance group. Audits are conducted on a biannual frequency in accordance to written procedures listed in paragraph 5.a. of this report. Quality Assurance Audit, Health Physics QAA-602-81-01, April 20-28, 1981, were reviewed for scope and timely response to the deficiencies identified. The NRC inspectors did not identify any problems in this area.

No violations or deviations were identified.

c. Radiation Protection Training

The NRC inspectors discussed initial and refresher radiation worker training with the training supervisor. The present training program appears to satisy the requirements of 10 CFR Part 19.12; however, did not include all recommendations of Regulatory Guides 8.27 and 8.29. The licensee stated Regulatory Guide 8.27 (dated March 1981) is titled, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants," and they are a high temperature gas cooled plant and not applicable to them. The NRC inspectors referred them to Section D, Implementation, which states that, "In the case of training programs at operating reactors, appropriate modifications to such programs should be made consistent with this guide as soon as practicable and no later than one year after publication of this guide." The NRC inspectors considered the licensee's facility an operating reactor and, therefore, were required to comply with these recommendations.

The licensee stated that they would review their training program against the recommendations of Regulatory Guide 8.27, in addition to the Institute of Nuclear Power Operations (INPO) which has published a proposed standard radiation worker training program. The licensee had planned to conform to the INPO training program and was scheduled to attend a meeting in mid-September on this program. This is considered to be an unresolved item (267/8221-06) pending revision of the training program to meet the recommendations of Regulatory Guide 8.27.

The NRC inspectors reviewed selected training records for new employees, regular plant staff, and health physics technicians. This review indicated requirements of 10 CFR Part 19.12 and the Station Training Program Administrative Manual were being met.

No violations or deviations were identified.

d. Radiation Protection Procedures

The NRC inspectors reviewed the licensee's procedures 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, and ANSI Standards, N13.1-1969, N13.11, N13.12, N18.1-1971, N18.7-1976, N322-1977, N323-1978, and N343-1978, and NUREG-0761.

The following procedures have been issued or revised since the previous radiation protection program inspection:

HPP-9, Establishing and Posting Controlled Areas, Issue 5

HPP-14, Analytical Instrumentation Room, Issue 11

HPP-19, Calibration of the Model 315 A-L Beckman CO Analyzer, Issue 4

HPP-20, Calibration of Radiation Detection Instruments, Issue 12

HPP-23, Receiving Radioactive Materials, Issue 6

HPP-27, Personnel Dosimetry, Issue 6

HPP-37, Emergency Kit Checklist, Issue 14

HPP-44, Radioactive Material Spill, Issue 2

HPP-46, Technical Specifications Related to Health Physics, Issue 2

HPP-48, Routine Maintenance, Inspection, and Cleaning of Respiratory Equipment, Issue 5

HPP-56, Reactor Building Exhaust Stack Discharge Activity Calculations, Issue 2

HPP-58, Calibration Procedure for Airflow Measuring Devices, Issue 2 HPP-60, Sampling Procedure for the Reactor Building Sump Effluent, Issue 1

HPP-61, Film Badge and Finger Ring Response Check, Issue 2

HPP-62, Portable Grab Sampler Operation Using 1260cc Marinelli Beaker, Issue 1

RCP-40, Operation and Calibration of the Whole Body Counting System, Issue 1

The NRC inspectors discussed these procedures with the radiation protection manager and noted where procedures were weak or inconsistent with plant operation. Procedure HPP-27, Section VI A.1, stated that personnel would receive a whole body count at the Colorado State Department of Health when terminating employment. However, the licensee had recently installed their own onsite whole body counting system and no longer used the Colorado State Department of Health system. All newly issued or revised procedures had been reviewed, approved, and issued in accordance with Station requirements.

No violations or deviations were identified.

e. Exposure Control

The NRC inspectors reviewed the station bioassay whole body counting operation and calibration program for agreement with the recommendations of ANSI N343-1978. The NRC inspectors discussed with the chief radiochemist, Procedure RCP-40, "Operation and Calibration of the Whole Body Counting System," and RCP-28, Routine Laboratory Functions. Procedure RCP-28 states the normal frequency for energy calibration check is weekly; the licensee performs the calibration check daily. ANSI-N343-1978, Section 15.3.3(3) states that these checks should be performed at least daily while the system is in use, and should be made at approximately 8-hour intervals. The licensee used radionuclides of Cr-51, Co-60, and Cs-137 for the body and lung calibration, and I-131 for thyroid calibration. Only one activity level of each radionuclide is used. ANSI-N343-1978, Section 15.2, recommends a series of measurements on various standard phantoms loaded with known quantities of radioactivity. These measurements shall be for the range of organ burdens of interest, i.e., 60-20,000 nanocuries of Co-60. The NRC inspectors inquired if any effort was being made to participate in an inter-calibration program with other facilities, as recommended in the standard. The licensee stated they would review the ANSI standard and also discuss this with the instrument vendor. This item is considered open (267/8221-01) and will be reviewed during a future inspection.

No violations or deviations were identified.

f. Posting, Labeling, and Control

The NRC inspectors, during a tour of the licensee's facilities on September 1-2, 1982, determined that the licensee was in compliance with the requirements of 10 CFR 20.203b, 20.203e, 20.203f, 20.207, and station procedures for the posting, labeling, and control of radioactive material and radiation areas. No high radiation areas or airborne radioactivity areas were noted.

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

No violations or deviations were identified.

7. NUREG-0737, "Classification of TMI Action Plan Requirements"

The NRC inspectors reviewed the licensee's progress and commitments 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 Effluent"

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 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(e) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, March 2, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(g) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(h) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(i) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(j) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(k) Memorandum, January 29, 1982, to file (USNRC R4) from T. F. Westerman (USNRC)

(l) Letter, March 19, 1982, to all Operating Nuclear Power Plants from D. G. Eisenhut (USNRC)

(m) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(n) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(o) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Waremboug (FSV)

(p) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Waremboug (FSV)

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

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

(s) Standard Review Plan, Section 6.4, "Habitability Systems"

(t) Calc - FSV Shielding Design Review for DBA - 1, Document No. C-70-002, September 23, 1980

(2) <u>Discussion</u>

An explanation of this item, per NUREG-0737, is given in the following:

"With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 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."

The licensee performed a design review which included a design basis accident where dose rates were calculated at various points in the plant.

(3) Conclusions

NUREG-0737 is written primarily for light water reactors which will not apply in every detail to the Fort St. Vrain High Temperature Gas Cooled Reactor. Therefore, the source terms in Regulatory Guides 1.3 and 1.4 are not applicable. Presently, Oak Ridge National Laboratory is performing a review which compares the source term calculations of the FSAR and the Gulf Atomic fuel model to determine the most conservative source term. The design review was done using the source term used in the FSAR.

During an accident situation, personnel would spend limited periods of time performing tasks in the reactor building. The design review gave dose rates that were acceptable to meet the General Design Criteria (GDC) to perform these tasks.

The control room peak gamma dose rate is less than 6 mR/h in an accident situation and this meets GDC 19 criteria for continuous occupancy.

The technical support center has a calculated dose rate of approximately 1 mrem/h.

The following plant systems, which require postaccident operation capability from the control room, were considered in the design review; reactor plant cooling water system, helium circulator auxiliary system, secondary coolant system, purification cooling water system, fire protection system, and Alternate Cooling Method.

The NRC inspectors determined that this item meets the conditions adequately as set forth in NUREG-0737, and should be considered closed.

No violations or deviations were identified.

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

(1) Documents Reviewed

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

(b) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(e) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, February 20, 1980, from F. E. Swart (FSV)

(g) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(h) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(i) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(j) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(k) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(l) Memorandum, January 29, 1982, to File (USNRC) from T. F. Westerman (USNRC)

(m) Letter, March 19, 1982, to all licensees of Operating Power Reactors from D. G. Eisenhut (USNRC)

(n) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(o) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(p) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(q) Letter, July 28, 1982, to R. A. Clark (USNRC) from D. W. Warembourg (FSV)

(r) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(s) FSV Radiochemistry Procedure - 34, 'Sample Handling and Log-In'

(t) FSV Health Physics Procedure - 14, 'Analytical Instrumentation Room'

(2) Discussion

Briefly, Item II.B.3 of NUREG-0737 requires the following:

"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 (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 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 (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. 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 or 1.4 release. The review should 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 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 and 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample within a shift)."

(3) Results

The design review, previously mentioned in paragraphs 7.a.(2) and (3), gives results whereby dose rates needed for this item (sample collection, transport, and analysis) to meet GDC-19 criteria is satisfied.

The NRC inspectors determined that samples of the reactor coolant and reactor building atmosphere could be collected in less than 1 hour. Also, the collection and analyses can be made in less than 3 hours. The licensee's computerized analysis system has a radioisotope library that is more than sufficient for the number of isotopes to be determined in an accident situation. In obtaining these samples, no auxiliary system has to be isolated.

In addition to the ability to obtain samples of the primary coolant, FSV has a continuous on-line sampler (RT 9301) that monitors primary coolant activity and provides a continuous indication of fuel degradation. Remote control room readout for this system provides a continuous indication of fuel integrity without the necessity of entering the reactor building.

Since FSV is a high temperature gas-cooled reactor, boron and chloride analyses during the accident are not applicable; hydrogen levels are determined with a gas chromatograph.

The radiochemistry laboratory, analyzing procedures, and equipment restricts background radiation levels to where sample analysis results will not contain objectable error. The Canberra Series 80 multi-channel analyzer with GeLi detectors are used in conjunction with a Digital Equipment Company PDP 11/44 computer to give the necessary accuracy, range, and sensitivity needed for isotopic determination. The offsite facilities of the State of Colorado Public Health and Colorado State University laboratories will be used as backup for sample analysis. The ventilation exhaust from the sampling station is filtered with charcoal adsorbers and HEPA filters.

The NRC inspectors had one area of concern. NUREG-0737 states that consideration should be given to:

Provisions for reducing plate out in sample lines, minimizing sample loss or distortion, and preventing blockage of sample lines by loose material, etc., in the sampling apparatus. These potential problems have not been investigated by the licensee. This item is considered open (267/8221-03) pending the licensee's investigation of the sampling system.

No violations or deviations were identified.

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

(1) Documents Reviewed

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

(b) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

(d) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart

(e) Letter, February 20, 1980, to S. A. Varga (USNRC) from F. E. Swart

(f) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(g) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg

(h) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg

(i) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(j) Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(k) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(2) Discussion

NUREG-0737 is written for Light Water Reactors (LWR) and states that the containment purge and vent isolation valves must close on a high radiation signal. To clarify this further, NUREG-0737 stipulates that these valves must be closed during operation of the reactor, and to implement this, 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.

At Fort St. Vrain (FSV) the "containment" consists of the Prestressed Concrete Reactor Vessel (PCRV) and the interspaces between the primary and secondary closures at PCRV penetrations. The "containment" pressure in the interspaces is always maintained above primary coolant pressure to ensure that no primary coolant helium can flow into "containment" if a leak develops in the primary coolant boundary, or into the environment if a leak develops in the secondary closure. The normal operating containment pressure is 710 psig and the normal reactor coolant pressure is about 5-15 psi lower. Also, the FSV reactor building is not considered to be containment and there is not any way to isolate it. The reactor building louver system releases to the environs for two minutes wherever the pressure in the building increases to 2.5 inches of water.

(3) Conclusions

The design of Fort St. Vrain (FSV) does not require provisions to purge and vent any secondary containment space, thus this item is only applicable to Light Water Reactors. Therefore, the NRC inspectors considers this closed.

No violations or deviations were identified.

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

(1) Attachment 1, "Noble Gas Effluent Monitor"

(a) Documents Reviewed

i. Letter, June 15, 1979, to G. Kuzmycz (USNRC) from D. W. Warembourg (FSV)

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

iii. Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

v. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vi. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vii. Letter, February 20, 1980, to S. A. Varga (USNRC) from F. E. Swart (FSV)

viii. Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

ix. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

x. Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

xi. Memorandum, January 29, 1982, to File from T. F. Westerman (USNRC)

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

xiii. Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

xiv. Letter, March 26, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

xv. Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

xvi. ANSI N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"

xvii. FSV Radio Chemistry Procedure 30, "Isotopic Calibration of Gaseous Activity Monitors"

xviii. SR 5.8.1 cd-Q, "Radioactive Gaseous Effluent System Calibration"

xix. FSV RERP-DOSE, "Offsite Dose Calculation Methodology"

xx. FSV Health Physics Procedure 56, "Reactor Building Exhaust Stack Discharge Activity"

(b) Discussion

NUREG-0737 position for this item is that the noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions. Multiple monitors are considered necessary to cover the ranges of interest.

Noble gas effluent monitors with an upper range capacity of E+05 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of E+05 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest.

Licensees shall provide continuous monitoring of high-level, postaccident releases of radioactive noble gases from the plant.

The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

Offline monitors are not required for the PWR secondary side main steam safety valve and dump valve discharge lines. Externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required.

Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

The design description shall include the following information:

System description, including:

instrumentation to be used, including range or senitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and,

the source of power to be used.

Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown is needed.

(c) Conclusions

Because FSV is a HTGR and the above NUREG-0737 position is for LWRs, some appropriate consideration should be given to this fact when reviewing the noble gas effluent monitors. In the meeting referenced in the memorandum of January 29, 1982, (7.d.(1)(a)(xi)), it was decided that the Oak Ridge National Laboratory (ORNL) will determine the upper limit for the Reactor Plant Ventilation Exhaust Stack monitor and that the upper limit values for this instrumentation should be based on the physical properties of the reactor instead of the fact that high level radiation monitors are commercially available. The NRC inspectors were unable to determine when ORNL would finish making the upper limit determination. Also, in the letter of March 24, 1982, (7.f.(1)(a)(viii.)) it was stated by the NRC that the licensee has met the intent of this item (Item II.F.1, Attachment 3) in a qualitative sense, and that the upper limit of the monitors specified in this item may be appropriate for LWR's only.

Due to this item's (II.F.1, Attachment 1 of NUREG-0737) stipulation that the noble gas effluent monitor must have an upper range capacity of E+05 uCi/cc, the licensee has met the intent of this requirement by designing an emergency stack monitor even though this is for light water reactors and not high temperature gas cooled reactors. The purpose of this monitor is to provide an estimate of noble gas activity released from the reactor building exhaust stack. This monitor consists of a lead shielded collimator (located on level 10 of the turbine building) and two portable radiation detection instruments, an Eberline E-500 detector with a GM probe and a ion chamber rate meter. Procedure HPP-56 describes how the readings from these instruments can be converted to exhaust concentration in uCi/cc. This system has a range of E-01 to E+05 uCi/cc.

The NRC inspectors determined that the maximum noble gas activity expected in the exhaust stack gas during an accident situation is 5E-02 uCi/cc which is approximately an order of magnitude below the maximum of the range (6.3 E-01 uCi/cc) of the Reactor Plant Ventilation Exhaust Stack monitor RT 7324-1 and approximately three orders of magnitude below the maximum of the range (1.5 E+01 uCi/cc) of radiation monitor RT 7324-2. The ranges of these inline monitors, RT 7324-1/2, are 9.5E-07 uCi/cc to 6.3E-01 uCi/cc and 2.3E-05 uCi/cc to 1.5E+01 uCi/cc, respectively, which give good range overlap and the necessary continuous range to cover normal operations (ALARA) through postaccident accident situations. These monitors are checked and calibrated on a monthly and quarterly schedule, respectively, according to RCP-30 and SR 5.8.1 cd-Q and are located on the turbine deck at elevation 4829 feet. Their readout modules (RIS 7324-1/2) and recorder (RR 93256) are calibrated on an annual basis and readout continuously in the control room.

The procedure to convert these monitor readings to release rates for offsite dose calculations is given in station procedure, RERP-DOSE, "Offsite Dose Calculation Methodology," Issue 1.

These monitor systems are on the essential power bus which provides uninterrupted power from the emergency diesel generators upon loss of normal power.

The licensee was unable to determine if these monitors were designed per ANSI N13.1 criteria. This item is considered open (267/8221-04) pending:

. the licensee's determination that monitors meet ANSI N13.1 criteria.

. the completion of ORNL Reactor Plant Ventilation Exhaust Stack monitor upper limit determination.

No violations or deviations were identified.

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

(1) Attachment 2, "Sampling and Analysis of Plant Effluents"

(a) Documents Reviewed

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

ii. Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

vi. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

vii. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

viii. Letter, March 20, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

ix. Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

x. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

xi. Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

xii. Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

xiii. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities"

xiv. SR 5.8.1cd-Q, "Radioactive Gaseous Effluent System Calibration"

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

xvi. FSV Health Physics Procedure - 53, "RT 7325-1 and RT 73437 Filter and Cart Removal (Emergency Accident Conditions)"

(b) Discussion

The clarifications for this item (Item II.F.1, Attachment 2) in NUREG-0737 states that the licensees shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates. Licensees shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition.

The sampling system design shall be such that plant personnel could remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the criteria of GDC 19 of 5 rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the NRC will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of ±20 percent. Further departure from the isokinetic condition need not be considered in design. Corrections for nonisokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969, may be considered on an ad hoc basis.

Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

(c) Conclusions

The particulate and iodine monitors continuously draw the effluent through a filter assembly and observe the radioactive buildup on the filter by means of a gamma scintillation detector. The paper-type (Whatman GF/A 47 mm) filter traps particles down to 0.3 micron with an efficiency greater than 95 percent. The filter is backed up a silver zeolite cartridge (RADeCo "Radioiodine Sampler" Model GY-130) which collects iodine in gaseous form with an efficiency greater than 90 percent. Both the filter and charcoal are monitored continuously by the gamma scintillation detector.

The plant gaseous effluents are sampled isokinetically for the above mentioned monitors (RT 7325-1 and 2) Monitor RT 7325-1 is used in conjunction with RT 7325-2 and both are located in the turbine building access bay on the north wall above the deaerator tanks at elevation 4921 feet. Monitor RT 7325-2 is a G-M detector which provides a high range capability for the system. Both of these monitors sample the reactor building ventilation exhaust and are read out in the control room on a multipoint strip chart recorder. These monitors have control actions of shutting down the turbine building ventilation system and placing the control room ventilation system on recirculation, wherever the setpoints are reached. These monitors are tested monthly and calibrated quarterly according to the procedures in SR 5.8.cd-Q.

The reactor building ventilation exhaust stack monitors (Eberline stack monitor, RT 73437-1, 2, and 3) monitor the effluent from the reactor building ventilation for beta particulate and iodine-131 radioactive contaminants. This monitoring system has isokinetic sampling with the same filters and collection efficiencies as previously dated for monitors RT 7325-1 and 2. It is comprised of two separate units. The detector and sampler unit located on EL 4912 feet of the turbine side and the readout unit is located in the control room. The system detectors are scintillation type detectors and their signals are sent to the readout unit in the control room where they are displayed. The readout consists of individual meter readouts, NORMAL, ALERT, and HIGH light indications, and a common chart recorder.

The licensee has the capability to remove, replace, and transport samples to the radiochemistry laboratory and meet the criteria of GDC-19 of 5 rem whole body and 75 rem dose equivalent to extremities during the duration of an accident. The procedure to perform this task is found in HPP-53. A shielding analysis study for transporting a loaded silver zeolite cartridge via a 2" thick lead pig determined that the unshielded cartridge has a contact dose equivalent rate of 20 mrem/h, and when contained in the pig the dose equivalent rate would be 1.3E-02 mrem/h at the surface.

The transported cartridges are analyzed in the radiochemistry laboratory outside the reactor building. A GeLi detector is used with a Canberra Series 80 Multichannel analyzer to determine the iodine content of the cartridge.

The vent stack airborne iodine concentration is continuously displayed, alarmed, and recorded in the control room. Two control room alarm functions are provided; the first being a trouble alarm on the iodine detector to indicate loss of background signal, loss of power, or an increased level of detected radiation above background but below the instrument setpoints, and the second being the high radiation alarm.

The NRC inspectors could not determine if any provisions had been made in the sampling systems to ensure that the adsorbers (zeolite cartridges) could not be degraded by entrained moisture in the effluent stream. The licensee had not considered this potential problem, therefore, this item is considered open (267/8221-05) pending the licensee's study of this item and the solving of the problem if any are found.

No violations or deviations were identified.

f. Item II.F.1, "Additonal 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 30, 1979, to all Operating Nuclear Power Plants from H. R. Denton

iii. Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

iv. Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

v. Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

vi. Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

vii. Letter, July 30, 1982 to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

viii. General Atomic Company Document Number C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

ix. SR 5.4.9-A3, "Area and Equipment Monitors Calibration"

(b) Discussion

For this item (Item II.F.1, Attachment 3), NUREG-0737 stipulates that the containment high-range radiation monitoring system must provide two radiation monitors in containment.

It specifies that the monitors have a maximum range of E+08 rad/h which includes 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 if it has upper range of E+07 R/h.

The monitors shall be located in containment(s) in a manner 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, and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV. Monitors that use thick shielding to increase the upper range will underestimate post-accident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gammas and are not acceptable.

The monitors must have the capability to detect and measure the radiation level within the reactor containment during and following an accident.

(c) Conclusions

Again, one must be reminded that the stipulation given for the high-range containment monitors are for light water reactors and FSV is a high temperature gas cooled reactor. The power density and fuel configuration are different for light water reactors and FSV. FSV's power density is lower and the fuel is encapsulated with a multilayered ceramic coating having a high temperature capability. This coating will delay the release of fission products after a reactor accident. Also, the prestressed concrete reactor vessel has a minimum thickness of nine feet. FSV does not have a containment building and the maximum gamma dose rate expected during a design basis accident is 1.4 rad/h in the reactor building. After 1000 hours into the accident, a maximum dose rate of 600 rad/h is expected from the main stack filters.

FSV is using the existing area radiation monitors (RT 93250, 93251, and 93252) to meet the requirements of NUREG-0737 containment high range radiation monitors.

These monitors are capable of reading up to 10 rad/h gamma dose rates. Although 10 rad/h is much less than E+07 rad/h specified for gamma radiation in NUREG-0737, FSV maintains that an appropriate radiation upper limit for their reactor building environment monitoring should be lower than that specified for light water reactors. An NRC letter (7.f.(1)(a) viii.) to FSV states that ORNL will determine the upper limits of the radiation level appropriate for the reactor building of FSV, and another letter (7.f.(1)(a) ix.) from FSV to the NRC states that a containment high radiation monitor is on order and should be installed by the end of 1982.

There are 17 area monitors located in the reactor building.

Each area monitor has a halogen-quenched G-M detector and an approximate sensitivity of 2 cps/mR/h. The monitors have a energy response of ±15% between 80 KeV and 2.5 MeV and a range of 0.1 mR/h to 10 R/h. The monitors are Gulf General Atomic Area and Equipment Monitor Detector Assemble RT-1. Each one has a local alarm (except for RT 93250-14) and the electronic equipment, recorders, and alarms are located in the control room.

The area monitors are tested on a weekly schedule and are calibrated quarterly according to SR 5.4.9-A3. They are source calibrated at 30-70 mR/h and 1.0 R/h. The area monitors are connected to essential power busses.

NUREG-0737 states that the containment high-range radiation monitor shall have the capability to detect and measure the radiation level within the reactor containment during and following an accident. The operating temperature limits are -58 to 167^{o}F for the FSV area monitors. In the FSV FSAR update, Figures 1.4-1, 2, and 4 show temperatures for accident situations in the reactor building that are greater than 167^{o}F for for periods of time up to 30 minutes. This would indicate that some of the area monitors would be inoperative under these conditions; therefore, these monitors would be unable to function properly continuously during an accident. This item is considered open (267/8221-07) until the licensee determines:

- During accident situations an adequate number of area monitors would be operating to determine radiation levels in the reactor building.

- Procedural changes and/or equipment modifications to be certain the accident could be "followed" by the area monitors even though more than one monitor is connected to an alarming annunciator.

- Installation of the ordered high range containment monitor.

No violations or deviations were identified.

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

(1) Documents Reviewed

(a) Letter, June 15, 1979, to G. Kuzmyca (USNRC) from D. W. Warembourg (FSV)

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

(c) Letter, October 29, 1979, to D. B. Vassallo (USNRC) from F. E. Swart (FSV)

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

(e) Letter, December 12, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(f) Letter, December 28, 1979, to S. A. Varga (USNRC) from F. E. Swart (FSV)

(g) Letter, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(h) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg

(i) Letter, December 30, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(j) Letter, August 6, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(k) Letter, August 26, 1981, to J. R. Miller (USNRC) from D. W. Warembourg (FSV)

(l) Letter, October 22, 1981, to S. J. Ball (ORNL) from D. W. Warembourg (FSV)

(m) Memorandum, January 28, 1982, to file (Region IV) from T. F. Westerman (USNRC)

(n) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(o) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(p) FSV Health Physics Procedure - 12, "Portable Air Sample Collection and Analysis"

(q) General Atomic Company, Document No. C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

(r) FSV Health Physics Procedure - 57, "Radiation and Airborne Radioactivity Monitoring During Abnormal Releases in the Plant"

(2) Discussion

The NUREG-0737 stipulates 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.

The effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

(a) The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

(b) Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

(c) Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

(d) The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

Each applicant and 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.

(3) Conclusions

For air sampling of radioiodines, the licensee uses a portable system weighing approximately 10 pounds that can be used in any area of the plant. This system includes a Radeco model H-809V air sampler with a Whatman GF/A filter and a Radeco silver zeolite "Radioiodine Sampler" model GY-130 cartridge, which has collection efficiency for iodine greater than 95 percent. These silver zeolite cartridges require no flushing with clean air or inert gases since they will not collect any of the noble fission gases.

The samples are collected for 5 minutes per procedures HPP-12 and -57, and taken to the radiochemistry laboratory for analysis on the multichannel analyzer with GeLi scintillation detectors previously described in this report. The analysis is performed according to procedure HPP-12.

The radiochemistry laboratory has a projected background dose rate of approximately 2 mrad/h from the reactor in an accident situation. The radiochemistry laboratory is on the ground level of the Technical Support Center which is outside of the reactor building and this complex has monitor RIT 7937 on the intake ventilation system. The high alarm setpoints on Monitor RIT 7937 are set to 3 E+04, 3 E+04, and 3 E+03 cpm for the gas, particulate, and iodine, respectively. These setpoints close the louvers routing the air through a filter system.

The NRC inspectors determined that the associated training for this item (Item III.D.3.3) could be improved to the extent that specific training for collection and analyzing of the iodine in emergency situations be given instead of relying upon the routine training in these areas. The added emphasis on the accident situation during specific training would be more beneficial. In addition to the routine training, the health physics and radiochemistry personnel participates in the two emergency drills annually where the necessary procedures are involved. The NRC inspectors inspected a sampling of the routine training and found it adequate.

The licensee also has two cart mounted iodine monitors (Eberline PING 1A) which has a single channel analyzer as part of each monitor. These monitors have very limited portability and are not easily moved to any position in the plant on a timely basis.

If needed, the licensee has a 2" lead pig, as previously mentioned in 7e.(3), to transport cartridges to the radiochemistry laboratory.

This item meets satisfactorily the intent of NUREG-0737 and should be considered closed.

No violations or deviations were identified.

h. Item II.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, March 30, 1980, to J. K. Fuller (FSV) from T. P. Speis (USNRC)

(d) Letter, December 20, 1980, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(e) Letter, August 6, 1981, to O. R. Lee (FSV) from J. R. Miller (USNRC)

(f) Letter, August 26, 1981, to J. R. Miller (USNRC) from D.W. Warembourg (FSV)

(g) Memorandum, January 29, 1982, to File from T. F. Westerman (USNRC)

(h) Letter, March 24, 1982, to D. W. Warembourg (FSV) from R. A. Clark (USNRC)

(i) Letter, June 1, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(j) Letter, June 10, 1982, to D. G. Eisenhut (USNRC) from D. W. Warembourg (FSV)

(k) Letter, July 30, 1982, to J. T. Collins (USNRC) from D. W. Warembourg (FSV)

(l) 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants - 19, Control Room"

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

(n) Standard Review Plan 2.2.3, "Evaluation of Potential Accidents"

(o) Standard Review Plan 6.4 "Habitability System"

(p) Regulatory Guide 1.78, "Assumptions for Evaluating

(q) Regulatory Guide 1.95, "Protection of Nuclear Power plant control room operators against an Accident Chlorine Release"

(r) General Atomic Company, Document No. C-70-002, "Calc-FSV Shielding Design Review for DBA-1.

(2) Discussion

In accordance with this item (NUREG-0743 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 shutdown under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 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.

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 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) Conclusions

In the various documents reviewed above, the licensee has made submittals to the NRC that provide a basis for their conclusion. In correspondence 19(a)(1)(g) and (h), it is implied that the licensee has met the requirements of this item, but a human factors study is needed. Also, correspondence 19(h)(1)(i) and (j) states that ORNL still has this item under review.

The NRC inspectors determined that the licensee's submittal addresses all the subjects entailed in this item (Item III.D.3.4) of NUREG-0737. Again, realizing that NUREG-0737 SRP 2.2.1-2.2.2, 2.2.3, 0.4, Regulatory Guides 1.78 and 1.95, respectively, for light water reactors and FSV is a high temperature gas-cooled reactor, it appears that the licensee has met the intent of these requirements. Therefore, this item (III.D.3.3) is considered closed.

No violations or deviations were identified.

8. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance, or deviations. The unresolved item disclosed during this inspection is discussed in paragraph 6.c.

9. Exit Interview

The NRC inspectors met with the licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on September 3, 1982. The NRC inspectors summarized the scope and findings of the inspection.