UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report No.: 50-62/78-2

Docket No.: 50-62

License No.: R-66

Licensee: University of Virginia
School of Engineering and Applied Sciences
Department of Nuclear Engineering
Charlottesville, Virginia

Facility Name: UVAR Facility

Inspection at: UVAR site, Charlottesville, Virginia

Inspection conducted: October 17-20, 1978

Inspector: S. C. Ewald

Reviewed by: _____ 11/28/78
A. F. Gibson, Chief Date
Radiation Support Section
Fuel Facility and Materials Safety Branch

Inspection Summary

Inspection on October 17-20, 1978: (Report No. 50-62/78-2)
Areas Inspected: Routine, unannounced inspection of (1) radiation protection including review of procedures, posting, labeling, personnel dosimetry, training, ventilation, radiological surveys, instrument calibrations; (2) waste disposal including review of reported effluents, liquid and gaseous effluent monitoring, sampling, and accountability, and solid waste disposal; (3) review of licensee actions relative to IE Circulars and Bulletins and reported events. The inspection involved 31 inspector-hours onsite by one NRC inspector.
Results: Of the eighteen areas inspected, no items of noncompliance or deviations were identified in fourteen areas. Two apparent items of noncompliance were identified in two areas (Infraction - failure to calibrate air monitor - paragraph 11; Deficiency - failure to maintain survey records - paragraph 7). Two apparent deviations were found in two areas (Deviation - failure to source and leak test pocket dosimeters - paragraph 8; Deviation - Laboratory hood flows less than 100 feet per minute - paragraph 13).

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DETAILS I Prepared by: _____ 11/28/78
S. C. Ewald, Radiation Specialist Date
Radiation Support Section
Fuel Facility and Materials Safety Branch

Dates of Inspection: October 17-20, 1978

Reviewed by: _____ 11/28/78
A. F. Gibson, Chief Date
Radiation Support Section
Fuel Facility and Materials Safety Branch

1. Persons Contacted

*T. G. Williamson, Director, Reactor Facility
*J. P. Farrar, Reactor Supervisor
*H. W. Berk, Radiation Safety Officer
*M. G. Bickel, Health Physicist
*B. L. Shriver, Research Associate
T. L. Porter, Senior Reactor Operator
G. E. Hedrick, Senior Reactor Operator
D. L. Andrews, Health Physicist
J. R. Gilchrist, Health Physicist

*Denotes those present at the exit interview.

2. Licensee Action on Previous Inspection Findings

(Closed) Open Item (77-3-1) Location of Air Monitor

Location of the constant air monitor is discussed in paragraph 12.

3. Unresolved Items

Unresolved items are matters about which more information is required to ascertain whether they are acceptable items, items of noncompliance, or deviations.

(78-02-05) Argon-41 Monitor Location. The inspector questioned the sensitivity of the reactor bay argon-41 monitor, as the result of ventilation flow measurements. This item is discussed further in paragraph 12.

(78-02-06) Sensitivity and Calibration of the Hand and Foot Monitor. Tests with radioactive sources of the hand and foot monitor indicate the system is not sensitive to contamination levels near licensee control limits. This item is discussed further in paragraph 9.

(78-02-07) Calibration of Contamination Monitoring Instruments. No routine program exited for the calibration of "friskers" used to screen contamination smears. This item is discussed further in paragraph 11.

4. IE Bulletins and Circulars

a. (Closed) Bulletin 78-07, "Protection afforded by Air Line Respirators and Supplied Air Hoods." Discussions with licensee representatives revealed respiratory protection devices are not used to limit or control the uptake of airborne radioactive materials by personnel. The inspector had no questions.

b. (Closed) Bulletin 78-08, "Radiation Levels from Fuel Element Transfer Tubes." The licensee does not have fuel transfer tubes as such. When spent fuel is moved (i.e., for shipment offsite) it is done in a shielded, single element transfer cask. The inspector had no questions.

c. (Closed) Circular 78-03, "Packaging Greater than Type A Quantities of Low Specific Activity (LSA) Radioactive Material for Transport." The inspector discussed the restrictions referenced in this circular with licensee representatives and reviewed recent shipping records. Solid radioactive waste handling is discussed further in paragraph 15. The inspector had no questions relative to the above circular.

5. Licensee Event Report

The licensee notified Region II on September 1, 1978, that low concentrations of xenon-135 and krypton-88 were found in a reactor pool water sample. The inspector reviewed the information supplied in the licensee's report of September 6, 1978 and discussed the event with licensee representatives. Licensee representatives stated they believed the isotopes were normally present but were undetectable prior to installation of a new demineralizer system due to interference of other activation products, specifically sodium-24. Due to the lack of any indication of other fission product isotopes, the licensee believes these gases do not indicate a fuel leak but rather represent diffusion effects in the fuel cladding. The event was reviewed by the Reactor Safety Committee on October 19, 1978. The inspector had no further questions.

6. Radiation Protection Procedures

The inspector reviewed various radiation protection procedures in effect including the UVAR "Standard Operating Procedures" (SOP), revised August 1978, and the University "Radiation Safety Guide." The inspector also reviewed separate procedures for: analysis of reactor pool and environmental water samples, calibration of the low background

beta counter, and calibration of area and portable monitors. The inspector had some questions concerning selected calibrations procedures and survey procedures. These questions are discussed in paragraphs 7, 9, and 11.

7. Posting, Labeling and Surveys

a. The inspector observed the posting of NRC Form 3's as required by 10 CFR 19.11 and noted the labeling and control of radioactive materials in accordance with 10 CFR 20.203 and 20.207. The inspector had no questions relative to these items.

b. The inspector performed independent direct radiation surveys with the reactor shutdown and at full power (2 megawatts). The inspector found no uposted radiation areas or high radiation areas and verified appropriate high radiation area access controls as required by 10 CFR 20.203(c). The inspector had no questions relative to the posting and control of radiation areas.

c. The inspector performed an independent survey for removable contamination. This survey identified two areas with removable contamination of 200 dpm/100 cm^2. Licensee control limits are 500 dpm/100 cm^2. The inspector noted the licensee technique for analyzing contamination smears involved initial screening with a thin-end window Geiger-Mueller (G.M.) tube and ratemeter. Licensee representatives stated if any activity is observed the smear is counted on the low background beta counter. Calibration of this instrument is discussed in paragraph 11.

d. The inspector reviewed survey procedures and requirements discussed in SOP 10.2 and Section 7.4 of the facility Safety Analysis Report (SAR). The SAR states that facility surveys will be performed on a weekly basis and the SOP requires daily surveys of areas when and where radioactive materials are handled. These daily surveys are the responsibility of the individual radiation worker and the inspector noted survey documentation sheets in two areas. The inspector noted that the SAR commitment of weekly surveys was not included in the survey procedures and commented that it might be appropriate to include it in the SOP.

e. The inspector reviewed weekly direct radiation and contamination survey records for the period January 6 through October 13, 1978. The inspector noted that no instances of contamination greater than 500 dpm/100 cm^2 had been revealed by the surveys reviewed. However, of the 41 weekly surveys performed in this period, records of 10 surveys were missing. Licensee representatives stated the surveys had been performed and the record apparently lost. The inspector noted all 10 instances occurred in the first

6 months of 1978. 10 CFR 20.201(b) requires records of surveys performed to evaluate radiation hazards be maintained. The inspector stated failure to maintain radiation survey records, as discussed above, was in noncompliance (78-02-02) with 10 CFR 20.401(b).

8. Personnel Dosimetry

a. The inspector reviewed the personnel dosimetry program and dose records. The dosimetry program involves several classifications of workers. Permanent facility staff are issued monthly film badge dosimeters incorporating beta, gamma, and neutron (NTA) sensitive emulsions. Non-routine workers (students in various classes) are issued quarterly film badge dosimeters with beta-gamma sensitivity only. Visitors are issued self reading pocket dosimeters. Pocket dosimeters are also used to supplement film badge dosimeters for all workers. The inspector reviewed monthly dose records for the period July, 1977 through August 1978 and quarterly records for the period July, 1977 through 1978. This review of records revealed no overexposures or exposures in excess of anticipated doses.

b. The film badge dosimetry used is supplied and read by a vendor. The inspector discussed the use of vendor dosimetry and various quality assurance programs that might be considered. The inspector expressed concern at the use of NTA film for neutron dosimetry without some form of quality assurance due to experienced fading problems and energy response of this film. Licensee representatives agreed to consider developing a routine dosimeter "spiking" program where one badge would be exposed to controlled doses of beta, gamma, and neutron radiations. The inspector stated the results of this review would be examined during a subsequent inspection (78-02-08).

c. The inspector discussed use of pocket dosimeters for supplementing vendor supplied dosimetry with licensee representatives. Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters," describes elements of an acceptable program and references ANSI N13.5, 1972, "Performance Specifications for Direct Reading and Indirect Reading Pocket Dosimeters for X- and Gamma Radiation." Regulatory Guide 8.4 Section C.1 states, in part, that pocket dosimeters should be tested for response and leak rate at maximum testing intervals of 6 months if the dosimeters are used to supplement another, primary, form of dosimetry. Acceptance-rejection criteria are referenced to ANSI N13.4-1972 Section 9. The inspector questioned licensed representatives about response and leak testing of their pocket dosimeters. Licensee representatives stated this testing is not performed routinely and had not been done for several years. This inspector stated that

failure to perform these tests deviated (78-02-03) from generally accepted industry practice as defined by Regulatory Guide 8.4 and ANSI N13.5-1972. Licensee representatives acknowledged the inspectors comments and agreed to develop a testing program.

9. Personnel Contamination Monitoring

a. Due to the nature of the licensee's operation, frequent handling of radioactive samples may be required of facility personnel. As a precaution, a "hand and foot" monitor is located near the main exit to allow staff to monitor themselves for contamination. University contamination limits of 500 dpm/100 cm^2 (beta-gamma on hands, skin and removable from shoes) are specified in the University Radiation Safety Guide.

b. The inspector questioned licensee representatives whether the instrument had been "calibrated" to verify sensitivity to control levels of contamination. Licensee representatives stated this had not been done but the monitor had identified several contamination problems in the past. The inspector performed response check of each channel in the monitor using reference beta and beta-gamma sources. These tests indicate the instrument had no sensitivity to beta particles and very low sensitivity to beta-gamma radiation (less than 0.1% efficiency). The poor sentivity apparently is the result of using thick walled (30 mg/cm^2) G.M. tubes. Licensee representatives stated use of thin window "pancake" tubes would be investigated to increase the monitors sensitivity. The inspector stated the monitor's sensitivity would be unresolved (78-02-06) pending the licensee's review and a review of regulatory requirements.

10. Training

The inspector reviewed training materials and recent attendance sheets for the radiation safety orientation lecture given to new workers. Licensee representatives stated attendance at this orientation is prerequisite to a workers being issued any film badge dosimetry. Licensee representatives also stated copies of Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," are distributed to all females attending these orientation lectures. The inspector had no questions relative to the above items.

11. Instrument Calibrations

a. The inspector reviewed calibration procedures and records for portable instruments. The inspector reviewed calibration records for 1978 of direct radiation survey instruments and had no questions. These calibrations are performed every two months. The inspector

noted that prior to 1978 calibration data was frequently logged as "OK" however all calibrations performed this year had the actual data logged.

b. The licensee uses "frisking" devices (a G.M. tube, usually with a thin window, and a ratemeter readout) to routinely screen contamination smears. The inspector questioned licensee representatives about calibration of these instruments and was told no routine program exits at the present time. The inspector estimated the efficiency of the friskers used in the reactor building (see paragraph 7), using a reference beta source, to be about 8%. The licensee's criteria for counting a smear on the low background beta counter is if indicated activity exceeds twice background (typically background equals 25 cpm). Thus the minimum detectable activity is about 300 dpm, well below the control limit of 500 dpm. Based on the above data, the instrument appears operable for screening contamination smears. Similar comments apply to the friskers used to screen smears in the solid waste drumming area (see paragraph 15). The inspector stated that failure to routinely calibrate these instruments would be unresolved (78-02-07) pending a review of regulatory requirements. Licensee representatives acknowledged the inspectors comments and agreed a calibration program would be appropriate.

c. The inspector reviewed calibration data and procedures for the various fixed radiation monitors. Technical Specification 3.2 lists the required monitor channels and Technical Specification 4.4 requires these channels be calibrated at intervals not to exceed 8 months. The inspector reviewed calibration records for the period March, 1977 through September, 1978 relative to four fixed area monitors (Reactor Face, Reactor Bridge, Demineralizer Room, and Hot Cell). The inspector also reviewed calibration data subsequent to maintenance of these channels for the period May 1977 through April 1978. The inspector had no questions on the above items.

d. A review of calibration records relative to the Constant Air Monitor, listed in Technical Specification 3.2, revealed the "calibration" included: (1) ratemeter adjustment; (2) high voltage check; (3) particulate filter replacement; (4) qualitative response check to a radioactive source. These items are performed on a semi-annual basis and the inspector reviewed records of these tests for the period March, 1977 through May, 1978. Technical Specification 1.11 defines a calibration to be an "adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures." The inspector stated that failure to perform a quantitative response test with a reference radioactive source as part of the routine calibration, was not consistant

with the definition of a calibration. The inspector stated, therefore, the instrument was not being calibrated in accordance with Technical Specifications 4.4 and 1.11. The inspector discussed this item in detail with management representatives who questioned whether the instrument vendor specification data would serve as a calibration base. The inspector stated that, since the vendor data was not specific to the licensee's instrument, but rather to the model of instrument, this was not acceptable. The inspector stated this item would be considered noncompliance (78-02-01) with Technical Specification 4.4. The inspector discussed with licensee representatives what elements would be required for an acceptable calibration program and stated, since the monitor's function is to measure airborne particulate activity, a calibration would need to include a quantitative test using a reference radioactive source with a technical basis to relate this data to an particular airborne concentration of radio-particulates. Licensee representatives acknowledged the inspectors comments and stated a full calibration program would be developed.

e. The inspector reviewed calibration records for the duct argon-41 monitor performed March 1978. The inspector noted comments in the calibration records stating a full calibration, in the test duct with known argon-41 concentration, will be done every six months. The inspector had no question on this item.

f. The inspector reviewed setpoint determinations and tests for the fixed area monitors and the argon [illegible] monitors. Alarm points are verified on a daily basis as part [illegible] SOP 4.1, "Daily Check List." The inspector verified the duct and reactor bay argon monitor setpoints were below 2 X 10^{-5} µCi/cc as stated in the facility SAR Section 4.9. The inspector had no questions on setpoint calculations and checks for these monitors.

12. Air Monitor Locations

a. During a facility tour October 17, the inspector noted the Constant Air Monitor was located inside the control room with a long sampling hose (approximately 10 meters) across the top of the control room to the sample point. The sample line was made from a section of lawn and garden hose. Referencing ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities" and its appendices, the inspector calculated nominal sample line distances for 50% and 100% sample losses due to gravity deposition using the licensee's sample line diameter and flow rate. Depending on particle diameter and density, 100% losses could occur in sample lines, with the same diameter and flow rate, as short as 1.5 meters. 50% losses for a 10 meter line could be expected for density-diameter combinations ranging from 10 g/cc and 1µm to 2 g/cc and 10 µm. The inspector discussed

these results with licensee representatives and expressed concern relative to the monitor's operability. The inspector stated, without specific data on particulate size and density, an accurate estimate of line losses is impossible but the above calculations indicate the potential for significant losses to be present. Licensee representatives stated this monitor is used to detect, among other things, fuel element leaks and stated the monitor would be moved to the end of the reactor pool where the circulated reactor water is discharged. This would eliminate the need for any sample line as well as improve the monitors response due to ventilation flow patterns discussed below. The monitor was relocated on October 18 and the inspector had no further questions.

b. The inspector performed independent ventilation flow measurements, using a thermoanemometer, profiling nominal flow patterns in the reactor room. The basic flow pattern is from the entry door across the reactor pool and out the ceiling ventilation duct. Nominal flows ranged from 30 to 50 linear feet per minute (lfpm). The inspector noted the argon-41 monitor used in the reactor room was located on the periphery of the flow path, on the corner of the control room ceiling. The inspector expressed concern that, with the existing air flow pattern, the monitor is located mid to up stream of any any radioactive gases generated at the reactor pool or bridge. The monitor might, therefore, never respond or respond much too slowly, to the generation of argon-41 and other radioactive gases. The inspector stated that these measurements indicated a potential problem with the usefulness of the argon monitor as installed and this item would be unresolved (78-02-05) pending a review by the licensee and consideration of relocating the monitor.

13. Ventilation System

a. In addition to flow measurements discussed in paragraph 12, the inspector also measured average flows through laboratory fume hoods and estimated total flow (cubic feet per minute-cfm) thru the reactor room. National Bureau of Standards Handbook 92, "Safe Handling of Radioactive Materials" Section 4.2, discusses ventilation criteria for fume hoods. Section 4.2.a states, in part, "For handling low to moderate levels of radioactive materials, the average velocity through openings in the hood must be 100 fpm." For higher quantities of radioactive materials flow rates of 125 to 200 fpm are specified. Four laboratory fume hoods in the reactor facility are used in conjunction with radioactive materials or potentially contaminated materials. Measured flows of two hoods located on the mezzanine near the demineralizer room indicated average velocities of 38 and 40 fpm. These hoods are used for routine handling of radioactive samples produced in the reactor. Two fume hoods located in separate laboratories had

average measured flows of 54 and 79 fpm. The first of these latter two hoods is used to process liquid samples, normally with low activities. The hood with the flow rate of 79 fpm houses a fixed experimental apparatus utilizing radioisotopes (technetium-99).
Licensee representatives installed new hepa filters in the mezzaine hood exhaust duct to see if the low flows were the result of plugged filters. No increase in flow was observed. All the above measurements were taken with the hood in a normal use configuration. The inspector stated that the above hood flows deviated (78-02-04) from accepted industry criteria as defined by NBS Handbook 92. The Radiation Safety Officer commented that 100 fpm is also the University acceptance criteria for hood flows.

b. The inspector measured flows through the reactor room access door to estimate total reactor room ventilation flow. Section 4.9 of the facility SAR indicates an expected total flow of 7000 cfm. The inspector verified the only ventilation supply was via the access door and measured an average linear flow of 198 fpm. This indicates a total volume flow of 3784 cfm, significantly less than the SAR value. Licensee representatives stated the significance of this item would be reviewed and any necessary corrective actions taken. The inspector stated the results of the licensee's review would be examined during a future inspection (78-02-09).

c. The inspector reviewed ventilation system surveillance tests performed to verify isolation and confinement capability required by Technical Specification 4.6. This review included monthly tests from January, 1977 through October 1978, of confinement interlocks including door and damper operability. Records of semiannual inspections of seals and gaskets for the period April, 1977 through October, 1978 were also reviewed. The inspector had no questions.

14. Liquid Waste

a. The inspector reviewed procedures for analysis and release of radioactive liquids. The current program entails collection of liquids in one of two batch tanks and, after sampling tank contents and pond water, release with pond dilution water to a stream. In addition a sample of the combined effluent is analyzed. The inspector reviewed release records for the period April, 1977 through September, 1978 and had no questions.

b. Licensee representatives stated a significant reduction in released activity is expected due to the installation of a new demineralizer system for processing reactor pool water. The inspector was shown the new system, using non-regenerable resin, and reviewed

installation records. The inspector acknowledged licensee comments relative to liquid waste reduction but stated some difficulties might be encountered in the handling and disposal of depleted resins.

15. Solid Waste

a. Solid waste handling is coordinated on a University wide basis and includes low level and low volume quantities of materials generated by the reactor facility. The inspector toured the waste handling building and reviewed solid waste records for a shipment August 3, 1978 and records being assembled for a shipment scheduled for October 23, 1978.

b. Waste is collected from the various university facilities and packaged in 55 gallon drums for shipment as Low Specific Activity. Estimates of total activity per drum are made from separate activity slips submitted with each bag of waste. These slips and filled out by individual isotope workers and disposed activity is correlated with isotope inventory records. The drums are checked for removable activity using smears counted on a frisker in the drumming area. The inspector had no questions relative to solid waste handling and shipping.

16. Exit Interview

At the conclusion of the inspection on October 20, 1978 the inspector met with management representatives (denoted in paragraph 1). The inspector summarized the scope and findings of the inspection. Items discussed included two items of noncompliance, two deviations, and three unresolved items.