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

Report Nos.: 50-280/78-27 and 50-281/78-27

Docket Nos.: 50-280 and 50-281

License Nos.: DPR-32 and DPR-37

Licensee: Virginia Electric and Power Company
P. O. Box 26666
Richmond, Virginia 23261

Facility Name: Surry Units 1 and 2

Inspection at: Surry Power Station, Surry, Virginia

Inspection Conducted: September 25-29, 1978

Inspectors: S. C. Ewald
L. L. Jackson

Approved by: A F Gibson 11/1/78

A. F. Gibson, Chief
Radiation Support Section
Fuel Facility and Materials Safety Branch

Date

Inspection Summary

Inspection on September 25-29, 1978 (Report Nos. 50-280/78-27 and 50-281/78-27

Areas Inspected: Routine, unannounced inspection of previous items, reactor coolant chemistry control, solid radwaste control, gaseous effluents, process monitors, resin systems, and licensee actions relative to IE circulars and bulletins. The inspection involved 66 inspector-hours on-site by two (2) NRC inspectors.

Results: No items of noncompliance or deviations were disclosed.

7812110225

DETAILS I

Prepared by: [signature] Date
S. C. Ewald, Radiation Specialist
Radiation Support Section
Fuel Facility and Materials
Safety Branch

[signature] 11/1/78 Date
L. L. Jackson, Radiation Specialist
Radiation Support Section
Fuel Facility and Materials
Safety Branch

Dates of Inspection: September 25-29, 1978

Reviewed by: [signature] 11/1/78 Date
A. F. Gibson, Chief
Radiation Support Section
Fuel Facility and Materials
Safety Branch

1. Individuals Contacted

*T. L. Baucom, Station Manager
*R. M. Smith, Health Physics Supervisor
*W. A. Thornton, Chemistry Supervisor
*W. W. Cameron, Superintendent of Technical Services
*S. Sarver, System Health Physicist
*W. L. Stewart, Superintendent, Station Operations
L. A. Johnson, Supervisor Engineering Services
R. G. Smith III, Engineer
*M. Tower, Quality Assurance Supervisor
*E. P. DeWandel, Administrative Assistant
*F. L. Rentz, On-Site Quality Assurance Engineer
P. P. Nottingham, Health Physics Coordinator-SGRP
J. Dodson, Senior Health Physics Technician
H. F. McCallum, Senior Health Physics Technician
D. Densmore, Senior Health Physics Technician
M. R. Beckham, Senior Health Physics Technician
C. E. Folz, Health Physics Technician
J. Horhutz, Instrument Supervisor
W. Snoberger, Assistant Instrument Supervisor

*Denotes those attending Exit Interview.

2. Licensee Action on Previous Inspection Findings

a. (Closed) Unresolved Item (78-02-04) Monthly Respirator Checks

The inspector reviewed changes made July 3, 1978, to section 8.3 of the facility Radiation Protection Manual. At this time, checks of respirators are made (1) during each cleaning, (2) prior to each use (by the wearer), and (3) a monthly check of respirators ready for issue. The inspector concluded these checks would assure any issued respirator would have been checked at least once during the previous month. The inspector had no further questions.

b. (Closed) Unresolved Item (78-08-01) Biweekly Tool Surveys

The inspector reviewed recent tool surveys and requested licensee representatives conduct a redundant survey on September 26. The inspector had no further questions.

c. (Open) Open Item (78-08-03) Liquid Waste Monitor Setpoint

The inspector reviewed data correlating liquid waste monitor response, average gamma energy and concentration for liquid releases during the period March 1 through April 15, 1978. The inspector discussed the data and analyses with licensee representatives and concluded the data revealed no useful information, due primarily to a combination of monitor response and background levels. The inspector discussed with licensee representatives other tests and calibrations that might yield useful data. One test the licensee agreed to consider would involve removing the detector from the liquid waste line and measuring relative efficiencies to several mono-isotopic sources with a range of gamma energies. This data could then be normalized to an operational geometry using the results of a monitor calibration performed February 14, 1978, that used a composite source. The Health Physics Supervisor noted that the above test would require prohibiting any liquid releases during the test and, therefore, once an estimate of the time involved is made, would require scheduling in conjunction with plant operations.

d. (Closed) Open Item (78-14-01) Calibration of Neutron Survey Instruments

The inspector reviewed changes approved September 27, 1978, to Health Physics procedure HP-3.2-14 detailing the basis and acceptance criteria for instrument response to a reference neutron source. The inspector had no further questions.

e. <u>(Closed) Open Item (78-14-02) Calibration of Gas Proportional Counter</u>

The inspector discussed modifications in the methods used to calculate instrument efficiency. Discussions with the counting room coordinator revealed backscatter effects are accounted for, and recent calibration data indicates an efficiency of about 45 percent. The inspector had no further questions.

f. <u>(Closed) Open Item (78-14-03) Neutron Film Spiking Program</u>

The inspector reviewed the results of initial spiking data and noted reported doses correlated well with calculated exposures. The inspector also noted that, for the data reviewed, the exposures involved a fast spectrum of neutron energies. The inspector suggested using a thermalizing medium (i.e., paraffin) in conjunction with the fast neutron source to produce a film exposure with fast and thermal components. The Health Physics Supervisor agreed to incorporate this into the routine program. The inspector had no further questions.

g. <u>(Closed) Open Item (78-14-04) Steam Generator Tent Air-sampling</u>

The inspector discussed the frequency of steam generator tent air sampling during steam generator maintenance work. The licensee has increased the frequency of airsampling, such that a sample is taken prior to initial entry and once per shift after, than when people are working in the tents. The inspector reviewed air sample results from the June 1978 outage, and had no further questions.

3. <u>Unresolved Items</u>

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

(78-27-01) Estimating the curie content in waste packaged for shipment. A comparison of three estimates of the curie content of a 55 gallon drum (Surry's procedures, North Anna's procedure, and inspector estimates based on techniques used at other facilities) revealed estimates differing by a factor of 10. Surry's estimate was the lowest of the three. This item is duscussed further in paragraph 7.

(78-27-02) Man-Rem dose commitments from resin systems. Discussions with licensee representatives and a review of reported summary data for 1977 indicates an estimated average dose commitment of 4.8 rem/year for operators involved with contaminated resin transfer systems. This item is discussed in detail in paragraph 8.

4. Licensee Event Report (LER) No. 78-24-03L

This LER involved the failure of an alarm board in the liquid waste monitor channel on July 29, 1978. The inspector discussed the event with licensee representatives and reviewed release data to verify compliance with release limits specified in Technical Specification 3.11.A.1. The alarm failure appeared to be an isolated event and the inspector had no questions relative to corrective action or consideration of generic implications.

5. Health Physics Staff Organization

The Health Physics Supervisor discussed with the inspector a recent reorganization of the health physics staff into several functional areas. These areas are: plant health physics, dose control, counting room, and respiratory protection. Four senior health physics technicians have been appointed "coordinators", reporting to the Health Physics Supervisor for each of the above areas. In addition, a health physics technician has been appointed training coordinator to administer the training programs for the health physics staff.

6. Man-Rem Summary Data-1977

a. The inspector reviewed summary data for 1977 submitted as per Technical Specification 6.6(b)(3). The inspector noted that over 58 percent (1410 man-rem) of the station dose resulted from steam generator maintenance activities with most of this exposure going to subcontract employees. The inspector calculated per capita exposures for each of the five job functions and noted the two highest dose jobs were primary system maintenance (average dose of 5.6 rem/year) and waste disposal activities (average of 4.8 rem/year) performed by station personnel. Totals including non-station utility and subcontract employees show waste disposal with the highest average exposure (3.3 rem/year) of all job functions.

b. The inspector questioned licensee representatives about the basis for the waste disposal man-rem estimates and possible causes of the relatively high values. Licensee representatives stated the majority of the waste disposal doses were received in conjunction with contaminated resin handling activities. The systems for preparing resins for disposal are discussed in detail in paragraph 8. The dose control coordinator stated that exposure data for these operations were not available as the operators dose per shift might be spread over several job functions, one including waste disposal. The reported doses were, therefore, estimates only, and the dose control coordinator stated the reported doses might be in error (conservatively) by as much as 20 percent. The inspector discussed the feasibility of obtaining separate data for resin handling operations to accurately describe the yearly dose commitments involved with these operations, and licensee representatives agreed to segregate this portion of an operators dose as accurately as practicable. The inspector stated this data would be reviewed in conjunction with the ALARA review discussed in paragraph 8.

7. Solid Radioactive Wastes

a. The inspector observed the loading of two contaminated filters from a storage bunker to shielded (Type B) shipping casks. The inspector reviewed survey data (radiation level and contamination smears) and verified compliance with labeling and contamination requirements specified in 49 CFR 172.406, 172.504 and 173.397. The inspector also reviewed selected shipping records for 1978 and had no unresolved questions. The inspector requested to see the certificate of compliance for the Type B shipping casks and reviewed the user requirements with licensee representatives. The inspector pointed out that compliance with the requirements specified in a certificate of compliance is the responsibility of the user and that this should be documented in the shipping records.

b. The inspector noted the licensee has implemented a program requiring an independent, on-site review of all shipping records and activities (sending and receiving) by a Quality Assurance Engineer prior to actual receipt or dispatch of a shipment.

c. Much of the solid waste shipped off-site consists of compacted, low-level contaminated materials shipped as Low Specific Activity (LSA) in 55 gallon drums. The inspector discussed the restrictions on LSA shipments, referenced in IE Circular 78-03, with licensee representatives, and a review of selected shipping records revealed no items of noncompliance. The inspector did question the methods used to estimate total curies in a 55 gallon drum to verify compliance with Type A quantity curie limits and 10 CFR 71.11. Licensee procedures call for measurement of the radiation level at the drum midpoint and multiplication by a conversion factor to obtain curies. Licensee representatives stated the conversion factors were taken from naval shipyard procedures, but that these procedures were no longer on-site. The inspector calculated curie estimates for a drum reading 400 mrem/hr. at the midpoint using the Surry station conversion factors, conversion factors used at the North Anna station and conversion factors derived by the inspector from references used at several other reactor facilities ("Determination of the Curie Content of Packaged Radioactive Waste using Measured Dose Rate" by W. B. Bowman II and D. L. Swindle, Health Physics, Volume 31, 1976). The three estimates were, respectively, 32 mCi, 312 mCi, and 168 mCi. The inspector discussed the wide discrepancy in the above estimates and stated that the basis for Surry's conversion factors would need to be reviewed to resolve the discrepancy. The inspector stated this matter would be unresolved (78-27-01) pending a review and comparison of the basis for the three techniques used and a determination of the adequacy of the technique currently used at Surry Power Station.

8. <u>Contaminated Resin Handling</u>

a. Based on discussions with licensee representatives and a review of submitted man-rem data (see paragraph 6), the inspector determined that persons handling contaminated spent resins had higher doses than most other radiation workers on-site. The inspector discussed the procedures for resin handling and toured the decon building with licensee representatives. Doses associated with resin handling are the result of, primarily, two operations; (1) transfer of resins to shipping liners, and (2) change out and storage of resins used to process liquid waste.

b. The licensee's installed system for handling contaminated resins involves storage in a resin holdup tank to allow for some activity reduction via decay. These resins are then transferred to a shipping liner. The liner is lowered through a hatch to a pit in the decon building, and an operator climbs onto the liner to hook up the flexible resin transfer lines. After the transfer is complete, the operator, again, climbs down to remove the resin lines and cap the liner ports. Due to the location of the resin liner loading area, the operator must enter a locked high radiation area (fields greater than one rem/hr.) for access to the resin liner. The sources of the high radiation levels are a large fluid waste treatment tank (FWTT) immediately adjacent to the liner pit and residual resin remaining in the transfer lines. Radiation levels from the FWTT vary depending on the contents and the inspector noted several posted "hot spot" areas on the tank, one indicating radiation levels of 7 rem/hr. Based on discussions with licensee representatives and inspector observations, it would appear difficult to effectively shield the tank. The inspector observed several 90 degree bends and right angle T's in the resin transfer lines. These items, along with several other resin traps, make it difficult to effectively flush the resin transfer lines and thus, some resin remains in the lines contributing to the high radiation levels.

c. The licensee installed a demineralizer system for treatment of liquid waste involving six small (10 cubic feet) disposable resin liners interfaced to the liquid waste system. These liners are located in the decon building adjacent to the resin liner loading area discussed above. The disposable liners are filled with clean resin locally and are lifted into separate shielded cylinders for use. System connections are made with flexible lines utilizing "quick disconnect" couplings. When a resin liner is removed from service the liners are disconnected, capped and then lifted over to a corner of the area for storage. The close proximity handling required, and the adjacent storage of liners, combine to create material radiation levels in this area.

d. The inspector discussed with licensee representatives his concerns relative to the above operations and maintaining personnel doses as low as reasonably achievable (ALARA). The inspector's concerns focused on operations in high dose-rate and contamination level areas and possible means to reduce levels or time required in these areas. One particular job, filling 10 cubic foot resin liners with clean resin, could reasonably be performed in a low dose-rate area and the already filled liners subsequently moved. Licensee representatives stated this would require installing some equipment to allow the filled liners to be moved from a loading area to where an overhead crane would have access to them. Licensee representatives agreed to review the feasibility of installing this equipment. The inspector stated the use of engineering and other controls to maintain doses incumbant to resin operations ALARA would be unresolved (78-27-02), pending further study by the inspector and the licensee.

9. Effluent Monitoring

a. The inspector reviewed process instrument calibration procedures CAL-RM-001, 002, 044, 045 and setpoint calibration procedures CAL-RM-003 and 004. These calibration procedures apply to a given type of instrument channel (i.e., Log ratemeter with GM Tube) and are called up for a specific channel (i.e., RM-LW-108, Liquid Waste Monitor) by Periodic Test P.T.-26.3. The inspector had no questions relative to these procedures.

b. Recent experience at other licensed reactor facilities have identified two potential problems relative to process monitoring systems. One problem relates to abnormally high readout from iodine monitor systems due to noble gas holdup in charcoal filters. Licensee representatives stated continuous iodine monitors are not used and determinations of radio-iodine releases are made from laboratory analysis of charcoal samples. The second problem relates to saturation of monitor channels incorporating Geiger-Mueller (GM) tubes. Under high release or dose rate conditions, a GM tube will saturate and a conventional ratemeter system will indicate an incorrect, nonconservative readout. Licensee representatives stated their GM type monitor channels do not have circuitry to compensate for this effect, but agreed to review the feasibility of modifying existing systems to incorporate this protection. The inspector stated the results of this review would be examined during a subsequent inspection (78-27-04). The inspector's discussion focused on one particular channel, the condensor air ejector monitor. Licensee representatives stated that upon receipt of an alarm, the air ejector effluent is diverted to containment to limit any off-site releases.

10. Effluent Sampling

a. The inspector accompanied a licensee representative to observe the weekly sampling of the ventilation vent. The procedure consists of drawing gas and tritium samples and replacing the particulate filter and charcoal cartridge. The latter two serve as a composite sample for the previous week and are analyzed simultaneously and combined with measured vent flow rate. Nominal flows are $7x10^4$ cfm and a weekly average flow is calculated from daily readings taken by operations personnel. The inspector noted the flow readout in the control room is a panel meter with a full scale of $1x10^6$ cfm and minimum reading of $5x10^4$ cfm. The inspector questioned the meter accuracy in the low ranges and reviewed meter calibration data. The meter is calibrated at apparent flowrates well above the $7x10^4$ normal reading. Since the meter indication is directly folded into release calculations, the inspector requested the licensee to investigate the accuracy of the meter indications in the operating range or to consider use of a different range panel meter. Licensee representatives agreed to review the matter and the inspector stated the item would be examined during a subsequent inspection (78-27-05).

b. The inspector discussed sampling of the condenser air ejector effluent and was shown the sampling station. The inspector commented that, due to the construction of the sample point, reliable estimates of particulate releases would be difficult. However, the inspector recognized that the only particulates one would expect to see would be daughters of noble gas isotopes. The inspector had no further questions on this item.

11. Reactor Coolant Chemistry

a. Technical Specification 3.1.F specifies the maximum allowable reactor coolant concentrations for chlorides, fluorides and oxygen. An inspector reviewed the results of tests for chlorides, fluorides and oxygen, conducted on both units, for the period January 1, 1978, through September 26, 1978. In addition, the analytical procedures used in conducting these tests were reviewed. The inspector had no questions related to this area of reactor coolant chemistry.

b. Technical Specification 3.3.A.3 requires that the boron injection tanks (one per unit) contain a boron concentration equivalent to at least 11.5 percent to 13 percent (by weight) boric acid solution. An inspector reviewed records of boron analysis for the boron injection tanks for the period January 1, 1978, through September 26, 1978. No problems were identified in this area.

c. Current reactor coolant system specific activity limits are specified in Appendix A-1 to License No. DPR-32 (Unit 1) and Appendix A-1 to License No. DPR-37 (Unit 2). These Appendices, issued May 5, 1977 and April 1, 1977, respectively, continue in force as a result of successive Orders for Modification of License. Paragraph 3.1.D of Appendix A-1 (requirements are the same for both units) specifies specific activity limits for reactor coolant in terms of dose equivalent iodine 131 and as a function of the average energy per disintegration (E-BAR) for certain nuclides in the reactor coolant. E-BAR will be more explicitly defined in a later paragraph.

(1) An inspector reviewed the test results of dose equivalent iodine 131 analyses, for both units, for the period January 1, 1978 through September 26, 1978. There were no questions pertaining to dose equivalent iodine 131 determination.

(2) An inspector selected several time periods during which power history logs indicated a power change exceeding 15 percent of rated thermal power in one hour and verified that

the isotopic analysis for iodine was conducted as required by Table 3.1.D-1 of Appendix A-1. The inspector had no questions in this area.

(3) An inspector reviewed some of the recent E-BAR determinations and noted that E-BAR is still being computed based, in part, on the superceded definition of E-BAR contained in the Basis for Technical Specification 3.1.D dated March 17, 1972. By this definition, E-BAR is the average sum of the beta and gamma energies, in mev, per disintegration for significant nuclides with half-lives greater than 30 minutes. These nuclides make up at least 95 percent of the total activity in the reactor coolant. Appendix A-1, discussed in paragraph c. above, states that E-BAR shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration, in mev, for isotopes, other than iodines, with half-lives greater than 15 minutes, making up at least 95 percent of the total non-iodine activity in the coolant. An inspector discussed the determination of E-BAR with management representatives and determined that the actual plant E-BAR procedure requires that all nuclides detected be included regardless of the half life. Any nuclides with half lives between 15 minutes and 30 minutes would have been included in the E-BAR calculation if detected during the analysis. This procedure, plus the fact that iodines are still included in the calculation, would make the existing E-Bar calculation method more conservative than the method described by current technical specifications. A review of analytical data showed the plant to be operating well below the specific activity limits for the reactor coolant. A management representative stated at the exit meeting that personnel in the licensing division of VEPCO had been informed of the situation and were in the process of determining if the E-BAR procedure should be revised to reflect exactly the definition of E-BAR as presented in Appendix A-1. This will remain an Open Item (78-27-06).

(4) An additional question regarding E-BAR involved the processing of the reactor coolant sample in preparation for counting on the multichannel analyzer. The sample is first degassed, then two separate portions of the degassed sample are taken. One of these portions is processed through an anion exchange resin and the other portion is processed through a cation exchange resin. The effluents from the two resin columns are counted separately to reduce dead time and improve resolution on the multi-channel analyzer system.

The assumption has been made that the combined results of the two portions would represent all of the non-gaseous gamma emitting nuclides in the reactor coolant sample. This assumption is not valid since some of the corrosion products, such as cobalt 58, are present as suspended matter instead of being ionized. The plant's analytical data confirmed this fact, since the concentration of cobalt-58 was approximately the same in the effluents from both of the resin columns. Since the corrosion products appear to be largely in the form of suspended matter, it is possible that a portion of these nuclides is being removed from the sample by the filtering action of the resin columns and that only the suspended particles small enough to pass through the resin are being included in the analytical results. The Chemistry Supervisor stated that, although an official study had not been performed to determine if there was an appreciable loss due to the filtering action of the resin, he had compared the concentrations of some corrosion products, as determined by filtration, with the corrosion product determinations used in the E-BAR procedure, and had found the difference to be negligible. The Chemistry Supervisor agreed with the inspector, however, that this may not constitute a satisfactory evaluation. The Chemistry Supervisor stated that the use of ion exchange resins to separate different factions of the sample was instituted when samples were counted using a sodium iodide detector and this technique reduced counting dead time and improved resolution of the different gamma peaks in the spectrum. He stated that the plant now uses germanium detectors to analyze the samples and the lower efficiency but higher resolution afforded by the germanium detectors should allow the samples to be analyzed directly, thereby eliminating the use of the ion exchange resin. The inspector agreed that this would eliminate the question. The Chemistry Supervisor stated that he would evaluate the direct counting method and eliminate the use of the ion exchange resin if the direct counting method gives satisfactory results. This item will be reviewed as part of Open Item (78-27-06).

12. Gaseous Radioactive Effluent Releases

a. Technical Specification 3.11.B specifies the release rate limits for gaseous effluents. An inspector reviewed the procedures for controlling effluents from the process vent and reviewed the analytical data for a limited number of process vent samples. There were no questions in this area.

b. An inspector reviewed the analytical data for a number of samples representing condenser air ejector discharges. Nuclides resulting from a steam generator tube leak were being detected, however, release rates were well within prescribed limits.

c. The specific control procedures and analytical results for all gaseous effluent release points were not reviewed during this inspection; however, the system of controls appears to be adequate for ensuring that release rate limits are met.

13. IE Bulletins and Circulars

a. Bulletin 78-07, "Protection Afforded by Airline Respirators and Supplied Air Hoods"

The inspector reviewed the licensee's response of August 14 and discussed the bulletin and related Los Alamos reports with the Health Physics Supervisor and the respiratory protection coordinator. The inspector noted the licensee does not use demand-mode airline respirators and has no plans to use them in the future. The inspector discussed tests made to ensure adequate air flow with supplied air hoods. The licensee first calibrated the pressure gauge on the distribution manifold and then measured air flow delivered to the hood at manifold pressures of 15 psig (minimum recommended) and 22 psig. The air flows were, respectively, 6 cfm and 8-10 cfm. Supplied air hoods are used with manifold pressures of 22 psig and, therefore, meeting the minimum criteria of 6 cfm appears to be no problem. Licensee representatives stated the pressure gauges are now calibrated every six months to assure accuracy. The inspector had no further questions.

b. Bulletin 78-08, "Radiation Levels from Fuel Element Transfer Tubes"

The inspector had requested the licensee perform surveys during the Unit 1 refueling outage in May 1978 (see RII Report Nos. 50-280/78-14 and 50-281/78-14, paragraph 5). The inspector reviewed survey results of May 11, 1978, revealing high levels (5 to 200 rem/hr.) at the gaps around the transfer tube shield blocks inside containment and levels of 0.5 to 12 rem/hr from "shine" at the fuel building -containment wall. The inspector reviewed the licensee's response to the bulletin of August 14, 1978, which stated these areas would be posted and barricaded, as required. The inspector suggested consideration also be given to the use of permanent or temporary shielding to reduce the radiation to acceptable levels. License representatives stated this possibility would be reviewed. The inspector stated that the results of surveys, with respect to Unit 2 and licensee precautions, would be reviewed during subsequent inspections (78-27-03).

c. 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 wastes are discussed further in paragraph 7. The inspector had no questions relative to the above circular.

14. Exit Interview

At the conclusion of the inspection, the inspector met with management representatives (denoted in paragraph 1). The inspector summarized the scope and findings of the inspection. Items discussed included two unresolved items and the status of previously identified items.