UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

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Report Nos.: 50-280/91-11 and 50-281/91-11

Licensee: Virginia Electric and Power Company
Glen Allen, VA 23060

Docket Nos.: 50-280 and 50-281 License Nos.: DPR-32 and DPR-37

Facility Name: Surry 1 and 2

Inspection Conducted: April 22-26, May 20-24, and June 5, 6 and 14, 1991

Inspector: J R Decker for 8/13/91
D. A. Seymour Date Signed

Approved by: J R Decker 8/13/91
T. R. Decker, Chief Date Signed
Radiological Effluents and Chemistry Section
Radiological Protection and Emergency Preparedness Branch
Division of Radiation Safety and Safeguards

SUMMARY

Scope:

This inspection was conducted to review the preoperational testing of the systems employed in Surry's new radwaste facility. This review included startup and preoperational procedures, training, the proposed process and area monitoring program, a review of installed or proposed instrumentation and equipment, and a determination of the progress the licensee had made in updating the Technical Specifications (TSs), the Final Safety Analysis Report (FSAR), and process and instrumentation diagrams. This inspection also included a review of the 1990 Radiological Environmental Report, the 1990 Semiannual Reports, and a confirmatory measurements inspection.

Results:

Inspector Followup Item (IFI) 50-280, -281/89-32-02, involving the performance of an evaluation of the maximum allowable backflow from fume hoods, was closed (Paragraph 2).

Surry Power Station has been involved in the construction of a Radwaste Facility. This facility will supplement the radwaste systems at Surry; by treating liquid and solid radwastes, providing a decontamination facility, and providing storage of packaged processed

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radwaste prior to shipment. The facility will have several radwaste processing systems, including: a Liquid Waste System; a Laundry Drain System; a Dry Active Waste System; a Spent Ion Exchange Handling System; and an Asphalt Solidification System. The facility will also include a hot machine shop and a radiochemical hot laboratory. Corporate Goals for the facility include: a reduction in the volume of radwaste shipped to disposal sites; a reduction in the release of radioactive materials released to the environment; a reduction in the radiation dose to plant personnel; reliable operation with state-of-the-art technologies; and an advanced computer controlled system for facility operation. The inspectors determined that the design and planned operation of the liquid, solid and gaseous waste systems in the Surry Radwaste Facility (SRF) were adequate for their intended purposes. The inspectors determined that the completion and operation of the SRF, as designed, would be a major component in the licensee's program to reduce the amount of radioactivity released in liquid effluents. The SRF was designed for efficient operation and maintenance. The inspectors considered the amount of planning and design, and the attention to detail, that was expended on this project, in order to minimize operator and facility personnel exposure, and to ensure efficient operation of the facility, to be a licensee strength (Paragraph 3).

The licensee was in agreement with accepted NRC values for the samples analyzed as part of the Confirmatory Measurement Program (Paragraph 4).

The inspector determined through a review of Count Room Quality Control, that the counting room detectors have exhibited generally stable performance (Paragraph 5).

The effluent releases and resultant doses were within TSs; 10 CFR 20, Appendix B; and 10 CFR 50, Appendix I limits. The total body and organ dose for 1990 was less than 2 percent of the 40 CFR 190 limit (25 millirem), while the thyroid dose for 1990 was less than 1 percent of 40 CFR 190 limit (75 millirem) (Paragraph 6).

There were not any significant radiological consequences attributable to the operation of the plant during 1990 from airborne, waterborne, aquatic, ingestion, or direct exposure pathways (Paragraph 7).

A review of a Quality Assurance Audit indicated that the audit findings and observations were of low safety significance. The inspector determined that the audit findings and observations had been effectively addressed, and included, where necessary, commitments by management to correct and prevent deficiencies. The audit was found to be well planned, thorough, and well documented. (Paragraph 8).

No violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*R. Bayer, Project Manager, Surry Radwaste Facility
*W. Benthall, Supervisor, Licensing
P. Blount, Supervisor, Radiation Analysis
*H. Collar, Supervisor, Quality Control
*D. Erickson, Superintendent, Radiation Protection
B. Garber, Technical Supervisor, Radiation Protection
*J. Hartka, Staff Engineer, Licensing
*M. Kansler, Station Manager
*L. Morris, Superintendent, Radiological Waste
*G. Price, Quality Specialist, Quality Control
*J. Price, Assistant Station Manager
*A. Royal, Supervisor, Radiation Training, Radiation Protection
*R. Saunders, Assistant Vice President, Nuclear Operations
*E. Smith Jr., Manager, Quality Assurance

Other licensee employees contacted during this inspection included engineers, mechanics, technicians, and administrative personnel.

Accompanying Personnel

*P. Stoddart, NRC Contractor

NRC Resident Inspector

*S. Tingen, Resident Inspector

*Attended exit interview

Acronyms and Initialisms used throughout this report are listed in the last paragraph.

2. Licensee Action on Previously Identified Inspector Follow-up Items (92701)

(Closed) IFI 50-280, -281/89-32-02: Performance of an evaluation of the maximum allowable backflow from fume hoods. Back-pressure problems and general degradation of the auxiliary building ventilation system had caused unmonitored leakage to the environment, and had caused reverse flow out of laboratory fume hoods, and into other areas outside the radiologically controlled area (portions of the Count Room, Hot Chemistry Laboratory, the Auxiliary Building, and the Service Building).

The inspector determined, through conversations with the licensee, that the back pressure problems in the ventilation system had been resolved. To correct the back flow problems, on May 8, 1990, the licensee had the honeycomb flow straightener in the Ventilation-Vent 2 stack cleaned of debris. This significantly helped correct the back pressure problem in the system.

Based on these conversations, and a walk down of selected portions of the ventilation system, the inspector determined that the licensee had corrected the problem, rendering the need for a formal evaluation of the maximum allowable backflow unnecessary. The inspector also determined that the ventilation system was being monitored to detect the development of future problems. This item is closed.

3. Surry Radioactive Waste Treatment Facility (84522, 84523, 84524, 84750, 86750)

Surry Power Station has been involved in the construction of a Radwaste Facility. This facility will supplement the radwaste systems at Surry; by treating liquid and solid radwastes, providing a decontamination facility, and providing storage of packaged processed radwaste prior to shipment. This facility is a "turn-key" venture, i.e. it is being built by a contractor and will eventually be turned over to Surry in a fully operational state, hopefully with most or all of the problems inherent in the startup of a new facility resolved. To ensure this, the contractor will run the plant for three years after operational testing is completed. (Surry personnel will be involved in the day to day supervision of the plant.)

The facility will have several radwaste processing systems, including: a Liquid Waste System; a Laundry Drain System; a Dry Active Waste System; a Spent Ion Exchange Handling System; and an Asphalt Solidification System. The facility will also include a hot machine shop and a radiochemical hot laboratory. Corporate goals for the facility include: a reduction in the volume of radwaste shipped to disposal sites; a reduction in the release of radioactive materials released to the environment; a reduction in the radiation dose to plant personnel; reliable operation with state-of-the-art technologies; and an advanced computer controlled system for facility operation.

At the time of this inspection, the licensee was involved in cold functional testing of specific components of the different systems, in particular, the evaporator, which is a major component of the Liquid Waste System. The nuclear industry's experience with evaporators has been dismal, and most, if not all, utilities have "mothballed" their evaporators and use demineralizers. However, Japan has had good experience with evaporators; and the contractor for the facility is a Japanese company. Surry has modeled their program after the Japanese, and they hope to reverse the trend set by the American nuclear industry with respect to the use of evaporators in radwaste. As a note, many different types of industries, worldwide, employ evaporators in similar functions successfully.

Following is a brief description of several of the components or processes used in the SRF. This information was obtained from interviews with the licensee, a review of documentation provided by the licensee, and from walkdowns. The inspector was aided in this review by an NRC contractor, hereafter referred to as an inspector.

a. Liquid Waste System (LWS)

The LWS was designed to receive, store and process liquid waste from Surry Power Station, and store and process liquid waste generated at the SRF. The LWS was comprised of three major subsystems;the Oil/Suspended Solids Remover, the Evaporator, and the Liquid Waste Filter Demineralizers. The system was designed to process 15,000 gallons per day.

(1) Collection Tanks

The LWS had a waste collection system comprised of two 25,000 gallon Liquid Waste Collection Tanks and two 25,000 gallon Liquid Waste Surge Tanks. Each of these tanks were fitted with an internal oil skimming system. In addition, the liquid would be treated by the SPI Oil/Suspended Solids Remover. The inspectors reviewed documentation provided by the licensee which recorded the calculations determining the minimum time for agitation or mixing to ensure homogeneous mixing and representative sampling of these tanks. In each case this calculated time was less than 30 minutes. The inspectors determined that the licensee planned on agitating these tanks for 30 minutes. This agitation time was included in the facility procedures, and was also computer controlled. The inspectors also determined that the licensee had performed tests with boric acid solutions which verified these numbers. The licensee had not performed these tests with insoluble material. This area will be reviewed further during subsequent inspections.

(2) The SPI Oil/Suspended Solids Remover

This system operates gravimetrically to remove oil and suspended solids. The removed oil would be collected in the Oil Drain Tank, while the suspended solids would be routed to the Solidification System or the Evaporator Bottoms Tank. The oil is fed from the Oil Drain Tank to the oil solidification area; there it might get processed, and/or shipped off site. The SPI Oil/Suspended Solids Remover was designed to remove oil up a concentration of to 10 parts per million (ppm). If there was oil in excess of this concentration, which would be determined by tank sampling prior to processing, the liquid would be routed through a Liquid Waste Filter.

(3) The Evaporator Subsystem

The liquid, after oil and suspended solids removal, would be sent to the Evaporator. The Evaporator Subsystem consists of a 30 gallon per minute forced circulation system using a mechanical vapor recompression system (MVR). In a MVR, system energy is conserved since the heat generated in the evaporator by condensing vapors is reused. These types of evaporators are more efficient in terms of production capability and economy of operation than the older technology multiple effect steam evaporators.

This process was designed to concentrate the feed or "liquor" to a concentration of 21,000 ppm boron or a total solids concentration of 25 percent by weight. When this has occurred, the concentrates generated would be gravity fed in batch mode to the Evaporator Bottoms Tank (5000 gallons). The bottoms would then be routed to the Bitumen Solidification System where they would undergo volume reduction, solidification and packaging (more detail is presented in Paragraph 3.b.(5)). The system for routing the evaporator bottoms was designed to prevent settling or accumulation of solids in the piping, valves, and pumps. The system also had the capability to be flushed out after transfer was completed.

Vapor from this process is passed through an entrainment separator and is compressed. The compressed superheated vapor is used to heat the incoming feed or "liquor," and is condensed. This condensed liquid is fed through a distillate demineralizer and stored in the Liquid Waste Monitor Tanks. This liquid may be reused in the SRF or discharged to the environment.

(4) The Liquid Waste Demineralizer Subsystem

The Liquid Waste Demineralizer Subsystem also operated at 60 gallons per minute. This system is comprised of five demineralizer vessels in series. The system was designed to allow the operator flexibility to optimize the use of the resin, which would minimize the generation of "secondary" waste. The licensee planned on running this system if the evaporator was not in operation. The inspector determined, through discussions with the licensee, that the demineralizer system and the evaporator could be run independently, in parallel, or in series. The Liquid Waste Demineralizer System has the same feeds as the evaporator, and would also discharge to the Liquid Waste Monitor Tanks.

b. Solid Radwaste Systems

The new Radwaste Facility at Surry Power Station provides systems for the processing of wet radioactive wastes into solid form and for the collection, separation, shredding and compaction of dry solid waste. The solid radwaste systems are designed to produce a product with minimum achievable volume as well as being acceptable for shipping and subsequent disposal at an offsite location.

(1) Dewatered Resins

Expended ion exchange resins will normally be slurried to High Integrity Containers (HICs). Contents of HICs will be dewatered using standard industry techniques; the maximum free liquid content for expended ion exchange resins in HICs (after dewatering) will be limited to less than 1 percent of the waste volume. If a HIC is not used, the maximum free liquid in the container will be limited to less than 0.5 percent of the waste volume.

(2) Spent Filter Elements

Filter elements are wound filter cartridges used in mechanical filters for the purpose of removing particulates from liquid waste streams. Prior to packaging as radioactive waste, spent filter elements will be allowed to drain dry to permit escape of water trapped in voids. Processing of spent filter elements will be based on the waste classification of the filters. Filters classified as Class A waste will not normally be encapsulated. Filters classified as Class B waste or Class C waste may be encapsulated in a manner described in the licensee's Process Control Program (PCP) (PCP; alternatively, un-encapsulated filters may be disposed of in a HIC, subject to the limitations set forth in the licensee's PCP.

(3) Dry Active Waste

Dry active waste will be manually sorted on a hooded controlled-air sorting table to separate tools and other reclaimable items form the waste stream and to separate the materials for drying prior to processing. From the sorting table, an enclosed conveyor belt system will carry dry waste material to a heavy-duty shredder machine, where the waste is shredded into bits or chunks with nominal dimensions of two inches or less. From the shredder, the waste is transported to a horizontal-ram-driven compactor. The licensee's tests have shown that a procedure employing approximately five successive high-pressure ram strokes produces a compaction volume with a minimum amount of spring-back, even with materials such as plastics. The licensee's ram compaction system provides a box-shaped product, which, when removed from the system, will be placed in a metal LSA box for eventual shipment to an offsite disposal facility. Except for the manual access hood, all components of the sorting, conveying and compaction system are fully enclosed and are exhausted to a HEPA-filtered ventilation exhaust system.

(4) Decontamination Facility

A Hot Machine Shop is provided for the maintenance of equipment and components of the power plant site. A separately ventilated decontamination facility and walk-in decontamination booth are located within the Hot Machine Shop area. The decontamination equipment for the decontamination facility and the walk-in decontamination booth includes a high-pressure liquid abrasive cleaner, "Turbulator", ultrasonic cleaner, and high-pressure freon spray cleaner; all of the above-named items are in common use at most nuclear plant sites. Hot Machine Shop ventilation air is exhausted through a HEPA filtered exhaust system.

At the time of the inspection, the licensee had been working with an engineering firm, TTI Engineering, in the field testing of a non-destructive decontamination process employing carbon dioxide as a decontamination medium. Extensive testing at the licensee's facility has shown this method to be highly successful in the decontamination of handtools, small parts, and even large mechanical components without apparent surface damage. The process employs solid carbon dioxide particles propelled by dry compressed air as the only cleaning medium. The carbon dioxide particles apparently break up on impact with surface of the material being cleaned and "flash" into carbon dioxide, with an attendant rapid volume expansion of about ten to one. The cleaning action is stated to be accomplished by the explosively expanding gas entering into the porous surface microstructure and flushing out the "foreign" contaminants. The surface being decontaminated is swept by a current of air to HEPA filters where airborne particulates are trapped. Larger foreign materials are lifted off of the surface by the flashing gas, fall to the floor of the decontamination chamber, and are vacuumed away to the HEPA filters. The advantages of the carbon dioxide decontamination system are: (1) little or no apparent damage is done to the item being decontaminated; (2) it is more effective than most existing decontamination methods; and (3) no secondary waste materials are added to the waste disposal stream. At the time of this inspection, the licensee was actively discussing the lease or purchase of the carbon dioxide decontamination system and was considering the feasibility of employing a similar system for the "in situ" decontamination of steam generators prior to future steam generator repairs.

(5) Asphalt Solidification System for Evaporator Bottoms and Other Radioactive Materials

The liquid waste processing system is described in detail in Paragraph 3.a. Bottoms from the vapor compression evaporator of the liquid waste processing system will be transferred from the concentrates tank of the liquid waste system to one of two batch tanks in the asphalt solidification system. From the batch tank, evaporator bottoms and preheated high-purity asphalt are separately piped to the mixing chamber of a thin film evaporator where they are mixed and heated. Water vapor is driven off from the mixture in the evaporator, condensed in the condenser and collected in a distillate tank. The remaining solids content of the evaporator bottoms is intimately mixed with the asphalt binder as it moves by gravity to the bottom of the thin film evaporator body. The mixture then flows out of the thin film evaporator into a 55 gallon DOT drum located beneath the evaporator. As a drum is filled, it is replaced by another drum and the filled drum is temporarily moved to a nearby location for cooling; as the drum cools, the asphalt mixture shrinks slightly from the top down, leaving a cone-shaped depression at the top of the drum, which results in the drum being less than 85% filled. To correct this condition, the drum will be later moved back under the evaporator and additional asphalt-solids mixture will be added to bring the drum contents to the desired level. After the drum has been filled the second time, it will be moved out from under the evaporator and capped using a remotely operated capping machine. The drum will then be allowed to cool and solidify prior to transfer to a drum storage area in the radwaste building.

The licensee's design for the asphalt solidification system also provides for its use in the solidification of spent resins. It should be noted that the licensee did not have plans in place to use the asphalt system for this purpose, but would have the capability of doing so in the event that the present system of disposing of resins in HICs should become unacceptable. In use as a solidification system for spent resins, the results would be slurried into the thin film evaporator, along with the asphalt binder, and the resin and asphalt would be mixed in the evaporator, water would be driven off and collected and the resin-asphalt mixture drained into drums in the manner described above for evaporator bottoms.

The solids content of the solidified product from the thin film evaporator of the asphalt solidification system is expected to be over 50 weight percent for evaporator bottoms concentrates and over 45 weight percent for resins. The solidified products are expected to be free-standing, liquid free, and monolithic in form, as specified in 10 CFR 61 and in the acceptance criteria of the disposal sites.

The inspector reviewed correspondence between the licensee's contractors and the NRC, and had telephone discussions with the licensee, regarding this process. The inspector determined that the licensee submitted a topical report to the NRC requesting approval to use this process at Surry Power Station (it should be noted that several facilities in the United States had previously received approval to stabilize their waste in this fashion). This report listed the results of structural stability tests performed on simulated samples of bitumen stabilized low level radioactive waste. These samples were two inch by two inch cylinders, much smaller than the actual product. The tests included a ninety-day soak test, and a compressive strength test. These tests were performed to demonstrate that the waste would remain stable under the compressive loads experienced after burial and upon exposure to moisture. The ninety-day soak test produced a one-half inch "skin effect," a change in the volume due to the uptake of moisture; and as a result of this uptake in moisture, substantial loss in compressive strength. The "skin effect" amounted to a large percentage of the sample volume; and if this was extrapolated to the full-sized waste form, it would pose the potential for compromising the structural stability of the waste. Based on these and other test results, the NRC ruled that the waste forms did not meet 10 CFR 61 criteria, and disapproved the interim use of the waste form.

The inspector determined that there was additional discussion and correspondence exchanged between the licensee's contractors and the NRC concerning this issue. The licensee's position was that these effects would not extrapolate linearly, and that a one-half inch "skin effect" would have a negligible impact on the structural stability of the full-sized waste form. The inspectors reviewed an interim agreement from the NRC, dated May 17, 1991, which allowed the licensee to proceed with the process pending completion of full-scale testing of the waste form.

The inspector determined during discussions with the licensee by telephone on August 9, 1991, that the licensee did not plan on full operation of the SRF until September 1991. The delay in the start of operations was due to the full-scale testing mentioned above. The licensee did not want to precede with radioactive waste materials prior to finding out the preliminary results of this testing, partially due to the difficulty in obtaining additional samples once "hot" operation has begun.

While experience with asphalt-based solidification systems in the United States to date has been less than successful, the licensee's engineering studies and the successful operation of asphalt-based systems in other countries have convinced the licensee's management that the system installed at Surry Nuclear Plant is practical. Licensee management also considers that the Quality Assurance (QA) programs now in place, coupled with sound engineering practices, good maintenance, adequate training of operating personnel, and provisions for a capable, proficient and dedicated staff will be adequate to assure the successful and continued operation of the asphalt solidification system.

On April 25, 1991, the inspector observed a preoperational test of the bitumen feed pump calibration. The bitumen feed pump is used to transfer bitumen from the bulk storage tank to the input section of the thin film evaporator for mixing with the input liquid waste stream. The procedure involved collection of two samples, each of two-minute duration, of flows of 25 percent, 50 percent, 75 percent, and 100 percent of design flow. The test was performed in a professional manner, all procedural steps were followed with precision, indicating either rehearsal or adequate pre-planning. All steps were satisfactorily completed and each step was documented. The inspector reviewed the preliminary results of this calibration and determined that the obtained calibration curve was linear. Based on this review, the inspectors determined that the calibration process was adequate for its intended purpose.

(6) Process Control Program (PCP)

The licensee maintains a PCP for the purpose of assuring that solidification of wet or liquid wastes meets the criteria of 10 CFR 61 and of the licensed disposal sites. At the time of the inspection, the licensee had prepared a revision of the existing PCP to accommodate the various solidification systems incorporated in the new radwaste facility. The inspector reviewed proposed Revision 1 to the station PCP, with the understanding that the proposed Revision 1 had not yet been approved by the Vice President - Nuclear Operations or by the NRC Office of Nuclear Reactor Regulation (ONRR); for this reason, the inspectors did not include an evaluation of the PCP in this report.

(7) Solid Waste Storage

The new radwaste facility at Surry Power Station is designed for the storage of up to a one year production of packaged solid wastes at the design capacity of the various solidification and processing systems. The facility includes materials handling equipment to facilitate the movement of waste during storage and during loading of transport vehicles for shipment to offsite disposal facilities. Also included is a radiation monitoring system for the monitoring of radiation levels of specific containers and of integrated radiation levels from multiple stored containers.

The inspector reviewed Generic Letter 81-38, "Storage of Low Level Radioactive Wastes at Power Reactor Sites," to determine its applicability at the SRF. This letter provides guidance concerning safety reviews and environmental assessments to licensees considering onsite contingency storage for up to five years duration. The inspector discussed this information with cognizant licensee personnel, and reviewed selected portions of the licensee's safety analysis for the SRF. Based on this selected review, the inspector determined that guidance and recommendations of the Generic Letter had been incorporated in the licensee's safety analysis. In addition, as noted above, this facility was not designed for the contingency storage of radwaste; the design of the facility provides for the storage of wastes which would be normally generated in one year of operation.

c. Gaseous Radioactive Waste Systems

The new radwaste facility provides for HEPA (High Efficiency Particulate Air) filter treatment of all potentially radioactive particulates in gaseous waste generated in the facility. All building exhaust ventilation from potentially contaminated zones is treated through HEPA filters. A number of isolated compartments within the radwaste building are exhausted through local HEPA filters before being vented through the building HEPA filter systems; these include the dry active waste sorting hood, hot machine shop, the carbon dioxide decontamination facility, and high pressure water spray decontamination facility. HEPA filter systems are provided with prefilters to remove large airborne particles and extend the service life of the HEPA filters.

Gases from all liquid radwaste tanks in the radwaste facility are vented through the tank vent system. Exhaust from the tank vent system passes through a separate treatment system consisting sequentially of a prefilter, HEAP filter, two-inch carbon bed, and downstream HEPA filter. This exhaust then is routed to the facility stack and monitored for radioactivity. The carbon bed is provided for retention of short-lived radioiodines which could be present in gases released from radioactive liquids which have not had sufficient time for decay of radioiodine.

As part of this inspection, the inspector reviewed the proposed changes to the licensee's TSs as a result of the SRF. These changes incorporated the location of the gaseous effluent release point associated with the SRF. This review included the licensee's analysis of the significant hazards consideration of the changes, and the Safety Evaluation performed by ONRR. The inspector determined, based on this review, that there would not be significant increases or changes in the amounts of effluents released offsite; and that there would not be a significant change in the amounts of individual or occupational exposure. ONRR concluded that this change to the TSs was acceptable. The location of the liquid effluent discharge point was not effected by the operation of the SRF.

d. Quality Assurance (QA) Program

The inspector reviewed the licensee's QA program for the new Radwaste Facility. The inspectors extensively discussed the workings of the QA program with the principal QA inspector assigned to the Radwaste Facility, reviewed the QA inspector's daily log, and reviewed the completed pre-operational test procedures for the facility's seal water system. It is noted at the time of the inspection, the seal water system pre-operational test procedures constituted the only available facility sub-system test package which had been totally completed, reviewed,and approved by licensee management. While other packages had been completed, they were in the stages of facility and corporate management review and approval and were not readily available. The inspectors also reviewed the seal water system piping and instrumentation diagrams (P&IDs) and additional vendor data on motors and pumps, The preoperational testing covered 24 items, including pumps, level indicators, coolers, filters and return flow lines. The inspectors noted only one minor administrative error regarding an entry of a cross reference to an identical but incorrectly identified procedural sub-section. Problems noted during the testing was satisfactory, and all items had completed the prescribed tests in a satisfactory manner.

e. The SRF Radiation Monitoring System (RMS)

As part of this inspection, the inspector also reviewed the RMS in the SRF. This review included a review of documentation, walkdowns of the system, and interviews with the licensee. This area was inspected to determine whether the licensee had a system sufficient to perform the surveys necessary to adequately evaluate the extent of radiation hazards, pursuant to 10 CFR 20.201(b).

The inspector determined that the Radiation Protection Department had performed several reviews and audits of the RMS. These reviews and audits were ongoing at the time of this inspection as the system was still being installed and/or modified. The inspectors discussed these findings and reviewed the resultant changes to the system. Primarily, these changes consisted of optimization of the monitor positions relative to proposed activities in the areas. These changes did include significant modifications to the SRF Vent Stack Monitor which would enable the acquisition of representative particulate and gaseous samples.

The inspector also determined that the monitors had local audible (siren) and visual alarms (flashing lights) on high radiation levels; as well as remote readouts in the SRF Control Room which would indicate the location and radiation level. These audits also included the liquid effluent discharge monitor, and the vent stack particulate and noble gas monitors. The liquid effluent monitor would automatically isolate the discharge on a high radiation alarm.

The inspector determined that the audits were comprehensive, resulting in several improvements to the system and optimization of the utility of the RMS.

f. SRF Operator Training

As part of this inspection, the inspector reviewed selected portions of the training program for the SRF operators. This review included observations and interviews of the operators at work in the Control Room of the SRF and in other areas of the plant. The inspectors also reviewed selected portions of training schedules, attendance records, examination summaries, operator evaluations of the classroom training, and the Operator Development Program Self Study Modules for the Chilled Water System and the Liquid Waste System. Based on this selected review, and based on observations and discussions with the operators, the inspector concluded that the training program for the SRF was effective, comprehensive and thorough.

g. The Safety Analysis for the SRF

The licensee performed a safety analysis as required by 10 CFR 50.59 to determine if the addition of the SRF involved an unreviewed safety question. The inspector reviewed selected portions of this document (NRF Document No.: C-20-122K-001, Rev. 1) which were germane to the areas of their inspection. The licensee concluded that the addition of the SRF did not pose any unreviewed safety questions. Some key conclusions which resulted from the analysis were:

(1) The SRF does not include, tie-in to, or indirectly affect any safety related equipment.

(2) The processes in the SRF had already been considered in the plant's Safety Analysis Report, with the exception of the Bitumen Solidification System. The Bitumen Solidification System has received interim approval from the NRC.

(3) The worse case hypothetical accident in the SRF would not result in a radiological release which is more limiting than those accidents currently described in the UFSAR. The licensee determined that the most significant release from an accident in the SRF would represent less than one percent of the maximum release associated with existing radwaste system accidents described in the UFSAR.

(4) Routine gaseous releases would be negligible, and routine liquid releases would be equal or better than those discussed in the UFSAR, in terms of radioactivity released.

(5) The SRF was designed and built to provide radiological protection to facility personnel. ALARA principles were incorporated.

The inspector determined, based on this selected review, that the operation of the SRF would not pose an unreviewed safety question.

h. Conclusions

The inspector determined that the design and planned operation of the liquid, solid and gaseous waste systems in the SRF were adequate for their intended purposes. The inspectors determined that the completion and operation of the SRF, as designed, would be a major component in the licensee's program to reduce the amount of radioactivity released in liquid effluents. The SRF was designed for efficient operation and maintenance. The inspector considered the amount of planning and design, and the attention to detail, that was expended on this project in order to minimize operator and facility personnel exposure, and to ensure efficient operation of the facility, to be a licensee strength.

No violations or deviations were identified.

4. Confirmatory Measurements (84750)

Pursuant to 10 CFR 20.201(b) this area was inspected to verify the licensee's ability to conduct precise and accurate measurements.

During this inspection, samples of reactor coolant and selected liquid and gaseous process streams were collected and the resultant sample matrices were analyzed for radionuclide concentrations using the licensee's counting laboratory and the NRC Region II mobile laboratory gamma-ray spectroscopy system. The purpose of these comparative measurements was to verify the licensee's capability to measure quantities of radionuclides accurately in various plant systems. Analyses were conducted using the licensee's three intrinsic germanium gamma spectroscopy systems. Sample types and counting geometries included the following: reactor coolant, 50-milliliter bottle; liquid waste, one-liter marinelli; containment atmosphere, 25-milliliter gas marinelli; and a charcoal cartridge. A particulate filter sample was generated for analysis by the filtration of 85 milliliters of reactor coolant. Comparison of licensee and NRC results are listed in Attachment 1, Table 1 with the acceptance criteria listed in Attachment 2. The results were in agreement for all sample types analyzed.

The inspector observed the licensee obtain the containment atmosphere sample. Proper sampling techniques and health physics practices were observed. The inspector reviewed selected portions of the applicable procedure. The portions reviewed were adequate for their intended purposes.

No violations or deviations were identified.

5. Count Room Quality Control (84750)

The licensee's count room Quality Assurance Program was reviewed to ensure compliance with selected and applicable portions of Regulatory Guide 4.15, Quality Assurance of Radiochemical Monitoring Programs (Normal Operations) Effluent streams and the Environment, Revision 1, February 1978. The following observations were made.

a. Quality control (QC) checks of the ND6600 gamma spectroscopy system detectors included a daily ten minute background check and source checks. The source checks were performed to determine if there were any changes to the resolution, efficiency and energy tolerance. The background, daily resolution check, and efficiency check were plotted and trended. The acceptance criteria for the these checks were plus or minus three standard deviations. The inspector reviewed current data (March 1991 to May 21, 1991) for the three detectors. This review indicated that the data was within acceptable limits, or appropriate actions had been taken when necessary. Detector stability was generally indicated.

b. Daily efficiency and background checks of the Beckman LS100C liquid scintillation counter used for tritium analyses were within specified control limits for May 1991, indicating general counter stability.

c. Daily background and efficiency checks of the PC-55 Gamma Products gas flow proportional counter used for alpha and beta analyses were within specified limits for May 1991, indicating general counter stability. The inspector also reviewed selected portions of the records for the last performed calibration.

The inspector determined through this selected review of Count Room QC, that the counting room detectors have exhibited generally stable performance.

No violations or deviations were identified.

6. Semiannual Radioactive Effluent Release Reports (84750)

TS 6.6.B.3 requires the licensee to submit a Semi-Annual Radiological Effluent Release Report, within the time periods specified in TS 6.B.3, covering the operation of the facility during the previous six months of operation. The inspector reviewed the semiannual radioactive effluent release reports for 1990. This review included an examination of the liquid and gaseous effluents for 1990 as compared to those of 1989 and 1988. This data is summarized below.

A comparison of liquid fission and activation products, gaseous fission and activation products, gaseous tritium, and gaseous particulate, for 1988, 1989, and 1990 showed no significant trends. There were some increases in specific streams (i.e. tritium, iodines) of effluents during 1990, as compared to 1988 and 1989, however it should be noted that the licensee had been in several outages during 1988 and 1989, and had been at power much of 1990. Also, as core life and plant run time increased, tritium and Iodine production increase. These increases were minor, and annual doses due to liquid and gaseous effluents varied insignificantly during this time period. The total body and organ dose for 1990 was less than 2 percent of the 40 CFR 190 limit (25 mrem), while the thyroid dose for 1990 was less than 1 percent of 40 CFR 190 limit (75 mrem).

The licensee had two monitors which were inoperable for longer than thirty days. One monitor, the Component Cooling Heat Exchanger Service Water Monitor 1-RM-SW-107C was scheduled for installation during the spring 1991 Unit 2 Refueling Outage. The installation of Component Cooling Heat Exchanger Service Water Monitor 1-RM-SW-107A, B, and D were completed. These monitors were designed to reduce the influence of biofouling. The operation of these monitors has been satisfactory. The second inoperable monitor was the Explosive Gas Monitor on the Waste Gas Holdup System. The licensee has installed new hydrogen and oxygen analyzers for this system. This work is completed and calibration and fine tuning of the system is on-going. Grab sampling was performed as required while these monitors were inoperable.

Radioactive Effluent Release Summary

	1990	1989	1988
No. of Unplanned Releases	0	0	0
Activity Released (curies)			
a. Liquid			
1. Fission and Activation Products	4.60+00	4.05E+00	2.41E+00

	2. Tritium	1.11E+03	4.29E+02	4.94E+02
	3. Gross Alpha	5.97E-05	6.98E-06	8.00E-05
b.	Gaseous			
	1. Fission and Activation Gases	4.50E+02	1.37E+02	3.66E+02
	2. Iodines	1.33E-03	3.89E-04	9.58E-03
	3. Tritium	2.17E+01	2.75E+01	2.79E+01
	4. Particulate	1.60E-03	1.99E-03	1.06E-02
c.	Volume of Liquid Wastes Released (prior to dilution) (liters)	1.74E+08	2.94E+09	2.58E+08

For 1990, Surry liquid and gaseous effluents were well within TSs, 10 CFR 20, and 10 CFR 50 effluent limitations.

No violations or deviations were identified.

7. Radiological Environmental Monitoring (84750)

TS 6.6.B.2 requires the submittal of a routine Radiological Environmental Operating Report. This report summarizes the results of the Radiological Environmental Surveillance Program; which measures accumulation of radioactivity in the environment, and determines whether the radioactivity detected is due to the operation of the Surry Plant. This program also assesses the doses to the offsite population due to plant effluents.

Pursuant to these requirements, the inspector reviewed this report for 1990.

a. Airborne Exposure Pathway

Airborne radioiodine results were all below the lower limit of detection (LLD), with no positive activity detected. Airborne gross beta results indicated no significant changes over the past five years, and compared favorably to the control location (quarterly averages for both the sample and control locations ranged from 14 to 19 picocuries per cubic meter). Only natural background radioactivity was detected in the samples analyzed for airborne gamma isotopic activity.

b. Waterborne Exposure Pathway

Of the gamma emitters, only naturally occurring potassium-40 was detected in any water samples. Iodine-131 was not detected. Tritium was detected in 12 of 24 composited samples, at an average value of 319 picocuries per liter, less than preoperational levels, and less than the average for the past five years. Surry discharge tritium levels were higher than the control location tritium levels (an average of 835 picocuries per liter compared to 475 picocuries per liter). Man-made or naturally occurring radioisotopes were not detected in well water samples. Tritium was not detected in well water samples.

c. Aquatic Exposure Pathway

River bottom silt samples were analyzed for gamma emitting radioisotopes. Cobalt-60, cesium-137 and cesium-134 levels have declined relative to the past five years trend. However there was an increase in 1990 values relative to 1989, due to increased operation of the plant.

The Radioactive Waste Treatment Facility is expected to reduce the volume and activity of liquid effluents; and reduce the impact of liquid effluents on the environment. Analyzes of shoreline sediment did not identify any radioisotopes attributable to the operation of the plant.

d. Ingestion Exposure Pathway

Iodine-131 and cesium-137 were not detected in the milk samples. Only naturally occurring radioisotopes were detected in the clam, oyster and crab samples. Cesium-137 was detected in one of four fish samples with an activity of 18.7 picocuries per kilogram, considerably below the reporting level of 2000 picocuries per kilogram.

Cesium-137 was detected in two out of five vegetation samples at an average of 11.3 picocuries per kilogram. this was less than the average concentration for the last five years. The required LLD for this measurement is 80 picocuries per kilogram, and the TS reporting limit is 2000 picocuries per kilogram.

e. Direct Radiation Exposure Pathway

The control and indicator thermoluminescent dosimeter (TLD) averages for 1990 indicated a decreasing trend in ambient radiation levels. Average values for the TLDs were from 4.2 to 8.0 milliRoentgen per month.

In conclusion, no significant radiological consequences to the environment were attributable to the operation of Surry in 1990 from airborne, waterborne, aquatic, ingestion, or direct exposure pathways.

No violations or deviations were identified.

8. Quality Assurance Audits (84750)

TSs 6.1.C.2.h.11, 12, and 13 require the Management Safety Review Committee to audit the Radiological Environmental Monitoring Program, the Offsite Dose Calculation Manual and implementing procedures, and the Process Control Program and implementing procedures, at least once every 12 months. The audit is performed in order to verify that these programs are being effectively implemented, and are in accordance with regulatory requirements.

Pursuant to these requirements, the inspector reviewed Quality Assurance Audit Report 91-03, which covered the aforementioned areas at Surry and North Anna Power Station. The audit was conducted from February 11, 1991 to March 21, 1991. The audit team was comprised of six personnel, including the team leader. The personnel were from both Surry and North Anna Power Station. The audit field investigation included; observation of activities, personnel interviews, area walkdowns, and procedure and document reviews.

The inspector reviewed the audit findings for Surry Power Station. There were two strengths, three findings, and six observations identified for Surry. The inspector discussed the audit findings and observations and the responses to these items with cognizant licensee personnel by telephone during the week of June 6, 1991. In general, the audit findings and observations were of low safety significance. The inspector determined, based on these phone conversations, that the audit findings and observations had been effectively addressed, and included, where necessary, commitments by management to correct and prevent deficiencies. The audit was found to be well planned, thorough, and well documented.

No violations or deviations were identified.

9. Exit Interview

The inspection scope and results were summarized on April 26 and May 24, 1991 with those persons indicated in Paragraph 1, and by telephone on June 5, 6, and 14, 1991, The inspector described the areas inspected and discussed in detail the inspection results as listed in the summary. No violations or deviations were identified. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

10. Acronyms and Initialisms

ALARA	As Low As Reasonably Achievable
CFR	Code of Federal Regulations
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container

IFI	Inspector Follow-up Item
LLD	Lower Limit of Detection
LSA	Low Specific Activity
LWS	Liquid Waste System
mrem	millirem
MVR	Mechanical Vapor Recompression System
NRC	Nuclear Regulatory Commission
ONRR	Office of Nuclear Reactor Regulation
P&ID	Piping and Instrumentation Diagrams
ppm	part per million
QA	Quality Assurance
QC	Quality Control
RMS	Radiation Monitoring System
SRF	Surry Radwaste Facility
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 1
TABLE 1

NRC-LICENSEE SAMPLE COMPARISON EVALUATION FOR
SURRY POWER STATION, May 20-24, 1991

Sample	Detector	Isotope	Concentration (uCi/unit) Licensee	NRC	Resolution	Ratio Licensee/NRC	Comparison
Reactor Coolant	#1	I-131	2.27 E-3	(1.52 ± 0.32) E-3	5	1.49	Agreement
		I-132	4.95 E-2	(5.59 ± 0.13) E-2	43	0.89	Agreement
	#2	I-131	2.10 E-3	(1.52 ± 0.32) E-3	5	1.38	Agreement
		I-132	5.02 E-2	(5.59 ± 0.13) E-3	43	0.90	Agreement
	#3	I-131	2.06 E-3	(1.52 ± 0.32) E-3	5	1.36	Agreement
		I-132	5.12 E-2	(5.59 ± 0.13) E-2	43	0.92	Agreement
Particulate Filter	#1	Cr-51	1.20 E-4	(9.11 ± 0.54) E-5	17	1.32	Agreement
		Co-58	8.99 E-5	(7.44 ± 0.18) E-5	41	1.2	Agreement
		Co-60	1.18 E-4	(1.35 ± 0.03) E-4	45	0.87	Agreement
	#2	Cr-51	1.11 E-4	(9.11 ± 0.54) E-5	17	1.22	Agreement
		Co-58	8.90 E-5	(7.44 ± 0.18) E-5	41	1.20	Agreement
		Co-60	1.21 E-4	(1.35 ± 0.03) E-4	45	0.90	Agreement
	#3	Cr-51	1.16 E-4	(9.11 ± 0.54) E-5	17	1.27	Agreement
		Co-58	9.44 E-5	(7.44 ± 0.18) E-5	41	1.27	Agreement
		Co-60	1.19 E-4	(1.35 ± 0.03) E-4	45	0.88	Agreement
Unit #1 containment atmosphere	#1	Xe-133	6.41 E-5	(5.96 ± 0.24) E-5	25	1.08	Agreement
		Xe-135	4.91 E-6	(4.36 ± 0.64) E-6	7	1.13	Agreement
	#2	Xe-133	6.51 E-5	(5.96 ± 0.24) E-5	25	1.09	Agreement
		Xe-135	4.46 E-6	(4.36 ± 0.64) E-6	7	1.02	Agreement
	#3	Xe-133	5.67 E-5	(5.96 ± 0.24) E-5	25	0.95	Agreement
		Xe-135	4.27 E-6	(4.36 ± 0.64) E-6	7	0.98	Agreement

Sample	Detector	Isotope	Concentration (uCi/unit) Licensee	NRC	Resolution	Ratio Licensee/NRC	Comparison
Charcoal Cartridge	#1	I-131	2.98 E-10	(3.30 ± 0.29) E-10	11	0.90	Agreement
		I-133	2.72 E-10	(2.58 ± 0.36) E-10	7	1.05	Agreement
	#2	I-131	3.19 E-10	(3.30 ± 0.29) E-10	11	0.97	Agreement
		I-133	2.49 E-10	(2.58 ± 0.36) E-10	7	0.96	Agreement
	#3	I-131	2.88 E-10	(3.30 ± 0.29) E-10	11	0.87	Agreement
		I-133	2.72 E-10	(2.58 ± 0.36) E-10	7	1.05	Agreement
Liquid Waste Test Tank	#1	Co-58	8.08 E-6	(8.15 ± 0.26) E-6	31	0.99	Agreement
		Co-60	1.73 E-5	(1.71 ± 0.04) E-5	43	1.01	Agreement
		Cs-137	3.79 E-6	(4.89 ± 0.24) E-6	20	0.78	Agreement
	#2	Co-58	8.08 E-6	(8.15 ± 0.26) E-6	31	0.99	Agreement
		Co-60	1.68 E-5	(1.71 ± 0.04) E-5	43	0.98	Agreement
		Cs-137	3.70 E-6	(4.89 ± 0.24) E-6	20	0.76	Agreement
	#3	Co-58	8.00 E-6	(8.15 ± 0.26) E-6	31	0.98	Agreement
		Co-60	1.67 E-5	(1.71 ± 0.04) E-5	43	0.98	Agreement
		Cs-137	4.00 E-6	(4.89 ± 0.24) E-6	20	0.82	Agreement

ATTACHMENT 2

CRITERIA FOR COMPARISONS OF ANALYTICAL MEASUREMENTS

This attachment provides criteria for the comparison of results of analytical radioactivity measurements. These criteria are based on empirical relationships which combine prior experience in comparing radioactivity analyses, the measurement of the statistically random process of radioactive emission, and the accuracy needs of this program.

In these criteria, the "Comparison Ratio Limits"[1] denoting agreement or disagreement between licensee and NRC results are variable. This variability is a function of the ratio of the NRC's analytical value relative to its associated statistical and analytical uncertainty, referred to in this program as "Resolution"[2].

For comparison purposes, a ratio between the licensee's analytical value and the NRC's analytical value is computed for each radionuclide present in a given sample. The computed ratios are then evaluated for agreement or disagreement based on "Resolution." The corresponding values for "Resolution" and the "Comparison Ratio Limits" are listed in the Table below. Ratio values which are either above or below the "Comparison Ratio Limits" are considered to be in disagreement, while ratio values within or encompassed by the "Comparison Ratio Limits" are considered to be in agreement.

TABLE

NRC Confirmatory Measurements Acceptance Criteria
Resolution vs. Comparison Ratio Limits

Resolution	Comparison Ratio Limits for Agreement
<4	0.4 - 2.5
4 - 7	0.5 - 2.0
8 - 15	0.6 - 1.66
16 - 50	0.75 - 1.33
51 - 200	0.80 - 1.25
>200	0.85 - 1.18

[1]Comparison Ratio = $\frac{\text{Licensee Value}}{\text{NRC Reference Value}}$

[2]Resolution = $\frac{\text{NRC Reference Value}}{\text{Associated Uncertainty}}$