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

JUL 29 1988

Report Nos.: 50-321/88-19 and 50-366/88-19

Licensee: Georgia Power Company
P. O. Box 4545
Atlanta, GA 30302

Docket Nos.: 50-321 and 50-366 License Nos.: DPR-57 and NPF-5

Facility Name: Hatch 1 and 2

Inspection Conducted: June 27 - July 1, 1988

Inspector: for P. Stoddart — Date Signed: 7/25/88

Approved by: J. B. Kahle, Section Chief — Date Signed: 7/25/88
Division of Radiation Safety and Safeguards

SUMMARY

Scope: This routine, unannounced inspection was conducted in the areas of liquid and gaseous radwaste management, liquid and gaseous effluent monitoring, reactor coolant chemistry and environmental monitoring.

Results: In the areas inspected, violations or deviations were not identified.

## REPORT DETAILS

1. Persons Contacted

Licensee Employees

B. C. Arnold, Chemistry Supervisor, Health Physics and Chemistry Department
*S. J. Bethay, Supervisor, Nuclear Safety and Compliance Department
G. Creighton, Program Coordinator, Nuclear Safety and Compliance Department
*O. M. Fraser, Manager, Site Quality Assurance
*V. A. McGowan, Chemistry Supervisor, Health Physics and Chemistry Department
A. H. Miller, Foreman, Health Physics and Chemistry Department
R. W. Ott, Supervisor, Training Department
*W. H. Rogers, Superintendent, Health Physics and Chemistry Department
*J. L. Shuman, Supervisor, Reactor Containment Systems
D. Smith, Superintendent, Health Physics and Chemistry Department
*R. W. Zavadoski, Manager, Health Physics and Chemistry Department

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

Nuclear Regulatory Commission

*J. Menning, Senior Resident Inspector
*R. Musser, Resident Inspector

*Attended exit interview

2. Audits (84723, 84724, 84725)

Technical Specification 6.5.2.8 (Units 1 and 2) required audits of unit activities to be performed under the cognizance of the Safety Review Board (SRB), encompassing the conformance of unit operations to provisions of the Technical Specifications and applicable license conditions at least once per 12 months. The inspector reviewed the following audit reports:

a. 87-SC-1, Quality Assurance of Site Chemistry, April 1-23, 1987

b. 87-ETS-1, Quality Assurance Audit of Environmental Technical Specifications, June 4-15, 1987

c. 87-SC-2, Quality Assurance Audit of Site Chemistry, August 18 - September 4, 1987

d. 87-RWC-3, Quality Assurance Audit of Radwaste Controls, October 19 - November 6, 1987

The inspector discussed audit results and reviewed followup actions for identified items with cognizant licensee representatives. Followup items were categorized as to relative significance and tracked by assigned identifying number. Due dates were established for responses and departmental responses were evaluated for completeness.

It was noted that the audit staff had been strengthened by assignment of additional industry-experienced specialists to the QA staff.

No violations or deviations were identified.

3. Procedures (84723, 84724, 84725)

Technical Specification 6.8.1 required written procedures to be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978; Process Control Program (PCP) implementation; and the Offsite Dose Calculation Manual (ODCM) implementation. The inspectors reviewed selected portions of procedures concerning effluent and reactor coolant sampling, process and effluent monitor calibrations, isotopic analyses, water quality analyses, in-place filter testing, analytical instrument calibration and radiological cross-check programs.

The inspector noted that several procedural rewrites and revisions were currently underway for the Chemistry Group. Procedures had been reviewed, updated, and approved in accordance with administrative control directives.

No violations or deviations were identified.

4. Liquid Radwaste Effluent Processing (84723)

The licensee's liquid radwaste processing system was demineralizer-based. The condensate cleanup system employed "body feed" demineralizers. This was a mode of operation in which a slurry of powdered demineralizer resin was precoated on porous tubes. As the initial "coat" became a less efficient ion exchange medium, an additional coat was applied during operation, in effect building-up a thicker medium and re-establishing the efficiency of cleanup. The process was then periodically repeated until space limitations prevented further coatings, at which time the system would be back-flushed to remove the resin, which was sluiced to a resin holding tank prior to processing for disposal.

The reactor water cleanup system (RWCS) used a mixed bed demineralizer to process a sidestream of reactor coolant from the reactor vessel in order to remove non-volatile impurities to limit buildup. In operation, resins were changed-out on an approximately 9-day cycle, with expended resins being sluiced to a resin holdup tank prior to processing for disposal.

The radwaste processing demineralizer system processed liquids from the liquid radwaste storage tank, from the floor drain storage tank, and from the condensate water storage tank. This system used a combination of precoat filter demineralizers and deep mixed bed demineralizers.

The radwaste processing demineralizer system was the only liquid radwaste cleanup system which processed plant liquids prior to disposal. The other systems - the RWCU and the condensate cleanup system - processed water for re-use in the reactor coolant system.

The inspector reviewed selected liquid radwaste discharge permits for the period of May - June 1988. Based on the analytical records which were part of the discharge permit packages, typical product sample analyses were in the E-06 to E-07 uCi/ml range, which indicated adequate performance of the demineralizer-based system and adequate control of process effluents.

The licensee's liquid effluent releases and the calculated dose consequences in the environment were less than the Technical Specification limits and less than the ALARA (as low as reasonably achievable) design objectives of 10 CFR 50, Appendix I. Based on the above, licensee releases were determined to be ALARA.

During 1986 and 1987, the licensee had experienced a small number of failed fuel assemblies and as a result had experienced higher-than-normal releases of gaseous iodine and noble gases from the reactor coolant (See Paragraph 10, Semi Annual Radiological Effluent Release Reports). The type of fuel defect apparently present in the Unit 2 fuel was considered by the licensee to be a result of small pinholes or hairline cracks in the fuel cladding. These defects were apparently of a type leading to the escape of noble gases and of volatile iodine species. However, any increase in fission and activation products as water-soluble or particulate contaminents was of a minor nature and did not contribute in a significant manner to the release of fission and activation products in plant liquid effluents in 1986 and 1987. The curie content of shipments of solid radwaste in 1987, which consisted largely of expended radwaste demineralizer resins, was slightly lower than shipments during the previous three years, providing added confirmation that the existence of fuel defects in the Unit 2 core did not add significantly to the curie content of processed liquid or solid radwaste streams.

No violations or deviations were identified.

5. Chemical And Radiochemical Determinations for Reactor Coolant Water (84723)

The inspector reviewed selected records of reactor coolant water analyses. The chemistry laboratory log books for March 1988 and April 1988, were reviewed in detail. Dose equivalent iodine (DEI) results, daily for the year 1987, as presented in the Annual Operating Report for 1987, were also reviewed.

The maximum value for DEI reported in 1987, was 0.131 uCi/ml, for Unit 2 on August 18, 1987; this was in the nature of a "spike" during a shutdown period when a small number of fuel defects was known to exist. Technical Specification limit for DEI was 1 uCi/ml for a period of 48 hours.

The March-April 1988 chemistry laboratory logs were also reviewed for determinations of pH, conductivity, chloride, and radioactivity concentrations.

No indications of out-of-specification results were noted.

No violations or deviations were identified.

6. Changes In Equipment And Procedures (84723, 84724)

The inspector reviewed the 10 CFR 50.59 evaluations reported in the 1987 Plant Annual Operating Report. Items reviewed included: 79-475, Rev. 1, and 79-476, Rev. 1, PASS sampling lines connected to the reactor pressure vessel and the primary containment; 81-132, Rev. 2, new sample lines for the hydrogen/oxygen analyzer and the post accident reactor coolant and containment atmosphere system; 85-105, replacement of the fission product monitoring system with the General Electric "NUMAC" system; and 87-013, connection of the "HI-HI" radiation signal and the "INOP" (inoperative) signal on the liquid radwaste effluent radiation monitor in series with the radwaste effluent isolation circuitry. All of the above changes were made to improve reliability of system performance.

No violations or deviations were identified.

7. Air Cleaning Systems (84724)

The Technical Specifications required leak testing of HEPA filters and charcoal adsorbers and methyl iodide charcoal retention testing for the air cleaning trains (2 per unit) of the control room environmental control systems (Units 1 and 2) and for the standby gas treatment system (SGTS) HEPA and charcoal filter trains (2 each for Units 1 and 2). Leak testing of HEPA and charcoal filter trains was required to be performed at least once every 18 months. Testing for methyl iodide retention by charcoal sample analysis was required to be performed at least once every 18 months or after 720 hours of operation, whichever came first.

Testing of air cleaning systems for leakage was performed by a licensee contractor, utilizing approved plant procedures with calibrated equipment provided by the contractor. HEPA filter leakage testing was performed with DOP aerosol and adsorber charcoal leak testing with a halogenated hydrocarbon aerosol. Methyl iodide testing of charcoal samples was performed by the contractor in a contractor laboratory using the procedures specified in the Technical Specification.

The inspector reviewed the test data and results of the standby gas treatment system (SGTS) filter train leak tests performed in March 1988, by the licensee's contractor. The tests were satisfactory and observed leakage was within the Technical Specification requirements. The inspector also reviewed the laboratory results of a methyl iodide retention test performed June 2, 1988, for a SGTS charcoal specimen; methyl iodide retention efficiency was calculated to be 99.114%, which was within the Technical Specification limit. The licensee's contractor was recognized by NRC as a qualified test laboratory.

No violations or deviations were identified.

8. Radioactive Effluent Monitoring And Instrumentation (84723, 84724)

The inspector reviewed selected procedures, logs, and effluent release permit packages. Determination of monitor alarm setpoints was clearly described and detailed in established procedures and in effluent release permit packages. Bases for setpoint determinations were clearly established in the procedures and affected operators and technicians appeared to have been adequately trained and qualified in the determination and use of setpoints.

Monitor readings correlated well with laboratory analysis results. The liquid radwaste effluent monitors were calibrated on a plant refueling outage frequency using a NBS - traceable liquid calibration source. Also during each liquid calibration operation, solid "transfer" sources were cross-calibrated to the liquid calibration source for use as check sources between calibration periods.

Liquid radwaste effluent monitors had local readouts in the radwaste control rooms and remote readouts in the reactor control rooms. The inspector verified that the Unit 2 monitor readouts in the radwaste control room and in the reactor control room showed good correlation.

The inspector reviewed logs of readings of the condenser air ejector pre-treatment and post-treatment process and effluent monitors for March and April 1988. Selected gaseous effluent release permit packages for May and June 1988 were also reviewed. The pre-treatment monitors were ionization chamber detectors with readouts calibrated in mR/hr; typical readings -- which reflected the presence of untreated short-lived noble gases -- were in the order of 50-200 mR/hr. Post-treatment gaseous effluent monitors were scintillation detectors reading on the order of 100-1,000 cps. Quantification of post-treatment gaseous releases was by laboratory analyses of monthly gaseous grab samples and average release flow rates over the period represented by the collected sample. Representative LLDs for the nuclides of interest were on the order of 2 E-06 to 6 E-08 uCi/ml.

Releases of noble gases from other release points were sampled, analyzed, and evaluated in a similar manner. Releases of radionuclides and particulates from all release points were calculated on the basis of

continous samples which were collected and analyzed weekly and from which composites were prepared quarterly for Sr-89 and Sr-90 analyses. Tritium samples were obtained monthly from each vent by a cold trap method and analyzed by an independent laboratory. Total releases for all nuclides were determined from analysis results and from release flow rates for each release point.

Dose rates and doses resulting from the release data were calculated using the methodology presented in the Hatch Offsite Dose Calculation Manual (ODCM) and compared to the doses listed in the Semi-Annual Effluent Release Reports. Manual calculation for gamma air dose due to Xe-133 in air and organ dose due to iodine-131 in milk, correlated closely with the doses listed in the Semi-Annual Effluent Release Report for 1987.

9. Confirmatory Measurements (84725)

The inspector reviewed the licensee's quality assurance program for verification of calibration of radioactivity identification and measurement laboratory instrumentation.

The licensee participated in quarterly cross-check programs with the Environmental Protection Agency (EPA) and with a qualified commercial analytical laboratory. The licensee was also in the process of establishing a third cross-check program with Plant Vogtle.

Typical cross-check samples from EPA included: gamma emitting radionuclides in water; iodine-131 in water; tritium in water; Strontium 89 and 90 in water; mixed beta-gamma emitters on filter paper; and iodine-131 on charcoal. Samples received each quarter from the commercial analytical lab consisted of four liquid samples containing mixed gamma, Strontium 89 and 90, tritium, and gross beta activity.

A typical quarterly list of analysis results included about 40 separate nuclide determinations performed on each of three detectors. A review of results for the four quarters of 1987, and the first quarter of 1988, indicated greater than 95% agreement for all nuclides. There appeared to be no specific pattern or bias in the disagreements, which indicated that such disagreements were of a random or statistical nature. In each case in which analyses which resulted in disagreement were re-analyzed, agreement was obtained.

On the basis of the above discussion, the licensee's quality assurance programs for verification of radioactivity identification and measurement laboratory instrumentation was considered to be adequate.

No violations or deviations were identified.

10. Semi-Annual Radiological Effluent Release Reports (84723, 84724)

Technical Specification 6.9.1.8 required the licensee to submit, within 60 days after January 1 and July 1 of each year, routine radiological effluent release reports covering plant operations during the previous six months of operation. The inspector reviewed the subject reports for January 1 - June 30, 1987, and for July 1 - December 31, 1987, during the inspection and discussed results with licensee representatives. The effluent release data summarized in the table below was obtained from the 1987 reports and from previous Semi-Annual Radiological Effluent Release Reports:

TABLE

EFFLUENT RELEASE SUMMARY FOR
PLANT E. I. HATCH, UNITS 1 AND 2

LIQUID RELEASES (Curies/Yr)

| Calendar Year | Fission and Activation Products | Tritium |
|---|---|---|
| 1984 | 1.32 | 102 |
| 1985 | 0.744 | 57.4 |
| 1986 | 0.790 | 28.5 |
| 1987 | 0.815 | 28.2 |

GASEOUS RELEASES (Curies/Yr)

| Calendar Year | Activation Products | Iodine | Tritium |
|---|---|---|---|
| 1984 | 12,600 | 0.101 | 33.2 |
| 1985 | 12,600 | 0.00599 | 26.6 |
| 1986 | 19,900 | 0.0235 | 33.4 |
| 1987 | 21,100 | 0.354 | 70.8 |

No violations or deviations were identified.

11. Environmental Monitoring (80721)

a. Contaminated Swamp

On December 3, 1986, a release of radioactive water resulted from the partial drainage of the spent fuel storage pools (SFSP). This water was determined to have drained into the onsite swamp east of the plant cooling towers. By letter of January 7, 1987, the licensee committed to an augmented radiological environmental monitoring program for the swamp area.

In a letter dated March 31, 1988, the licensee provided the results of the augmented radiological environmental monitoring program through 1987. The inspector reviewed the results provided in the March 31, 1988 letter.

External radiation levels were below detection limits for portable survey instrumentation and all monitoring program results were based on laboratory analysis of water, soil, and vegetation samples. The swamp lies on the flood plain of the Altamaha River and was inundated by high river water on two occasions during January and March (1987). The flood waters flowed through the swamp in an easterly direction, paralleling the flow of the river. Dilution of the swamp water by the flood waters accounted for initial reductions of tritium content of water samples for all sample locations but had mixed effects on nuclides other than tritium in soil and vegetation samples, resulting in decreases at some locations and little or no change at other locations, possibly indicating precipitation or uptake prior to the flushing effects of the flood waters.

Nuclides seen in samples of vegetation (grass) and soil ("muck") were typical of nuclear plant reactor system fluids, especially Mn-54, Co-60, Zn-65, Cs-134, and Cs-137.

During the first three months of the program, water and soil (mud or "muck" -- soil or mud samples contained a substantial fraction of root and other organic material and were subsequently referred to as "muck") samples were collected at two points at the edge of the swamp. During the second quarter, water sample activity levels had dropped to near-background levels but muck and grass samples contained "pronounced" levels of radionuclides and it was decided to expand the sampling program by adding four new locations for the May and June samples.

In July 1987, the perimeter of the swamp pond was examined on foot. It was discovered that at two locations, water was flowing freely from the pond nearest to the cooling towers further into the swamp, apparently as the result of earlier flooding which had breached some of the beaver dams which had created the pond. The sampling program was extended by adding three more sampling points -- one at each pond release point and one downstream. In August, six more sample locations were established along the eastern site property line and along the right back of Bay Creek near its confluence with the Altamaha River, about three-quarters of a mile east of the eastern site property line. Each of these six points would be along an effluent path through or from the swamp under flood conditions.

Positive results of all sample analyses were provided in tabular form in units of picocuries per liter for liquid samples, picocuries per kilogram (dryweight) for muck, and picocuries per kilogram (wet weight) for grass samples. Samples collected at "Point A", the sample location nearest to the cooling towers, generally dominated

those collected at other locations, both in the number of radionuclides detected and in the concentrations of these radionuclides. A high variability was seen in all samples taken from a given sample point on three or more occasions; the variability was less pronounced for grass samples than for muck samples.

10 CFR 20.105(b)(2) provided that a licensee may not possess, use, or transfer licensed material so as to create in an unrestricted area, from radioactive material and other sources of radiation in his possession, radiation levels which, if an individual were continuously present in the area, could result in that individual receiving a radiation dose in excess of 100 mrem in any seven consecutive days. One-hundred mrem in a period of seven days would be represented by a radiation dose rate of 595 microrem per hour.

The licensee used the guidance in NRC Regulatory Guide 1.109 to calculate the direct radiation dose rate to an individual standing on contaminated ground. The licensee calculated direct radiation dose rates based on various muck samples collected on September 15, 1987. The licensee also made field measurements on the same date, using a calibrated "Micro-R" radiation detection meter at a height of one meter above the spots on the ground where the samples were collected.

Table: Calculated and Measured Dose Rates

| Sample | Calculated Dose Rate Due to Deposited Material (Micro R/hour) | Measured Dose Rate (Including Natural Background) (Micro R/hour) |
|---|---|---|
| Point A | 0.736 | 12 |
| PL-2 | 0.117 | 8 |
| PL-3 | 0.261 | 13 |
| MBC | 0.086 | 11 |
| Upstream | 0.069 | 12.5 |

It was noted that the highest calculated dose rate for a muck sample collected from an unrestricted area (samples MBC and Upstream) was more than a factor of 2,000 below the 10 CFR 20.105(b)(2) limit of 100 millirem in any seven consecutive days (or 595 uR/hr), while the highest on-site sample (Sample A) was a factor of more than 800 below the 10 CFR 20, Appendix B, Table II, Column 2, MPC for an offsite sample location. The calculated dose rate for an upstream sample was determined on the basis of naturally-occurring radioactive materials but did not include factors such as the contribution of cosmic radiation and the presence of radon decay products in the atmosphere. The total naturally-occurring radiation level in the continental United States has been demonstrated to range from approximately 10 uR/hr at sea-level to approximately 100 uR/hr at an elevation of 5,000 feet; since the Plant Hatch elevation is about 70 feet above

sea level, the anticipated naturally-occurring radiation background level as seen by a calibrated Micro-R meter should average slightly over 10 uR/hr. The measured radiation levels shown in the table above were consistent with the expected natural radiation levels. As a result, the component of the measured dose rate due to the presence of the contamination could not be distinguished from the natural background level.

The licensee planned to continue the augumented monitoring program into calendar year 1988, and will report on the results for 1988, at a later date.

b. Tritium Contaminated Ground Water

Prior to 1985, the licensee reported quarterly to the NRC on the results of a groundwater tritium monitoring program. Elevated tritium levels in the groundwater on the Plant Hatch site were presumed to have originated from leakage or seepage from plant structures several years previously. The quarterly reports were terminated in 1985, although the licensee committed to continuing the program, with program records to be maintained for NRC review.

The inspector reviewed the results of the groundwater tritium monitoring program for the months of December 1987 through May 1988. Analysis results of samples from the groundwater monitoring wells ranged from the lower limit of detection of approximately 2.0 E02 pCi/l up to 8.46 E05 pCi/l. The observed values were not significantly different from those reported for the first quarter of 1985, which appeared to show that the groundwater condition was stable, with minimal lateral flow.

The 10 CFR Part 20, Appendix B, Table II, Column 2, value for offsite maximum permissible concentration of tritium effluents was 3 E06 pCi/l. On this basis, the maximum concentration observed in the groundwater was approximately 28% of the offsite MPC.

The groundwater wells having the highest concentrations of tritium were those immediately adjacent to the plant structures. Monitoring wells were spaced at various locations and distances within a distance of about 100 yards from the Unit 1 Reactor and Turbine Buildings, with a small number of wells near the Unit 2 Reactor and Turbine Buildings. All identified plant tritiated water leaks to date were associated with Unit 1. That the bulk of the tritium in the groundwater was associated with Unit 1 and that the tritium does not appear to be migrating was demonstrated by the analysis results, with samples from the wells immediately adjacent to the buildings showing the highest concentrations and wells at a distance of 50 to 100 yards showing background or near-background concentrations. The licensee's sample results appeared to demonstrate that the tritium

contaminant was not increasing or decreasing and was not migrating by either lateral movement or dispersion by diffusion. With little or no movement or dilution occurring in a three year time frame, the potential for the tritium to reach potable water sources at a distance of several miles was considered to be minimal.

c. Environmental Report

The inspector also reviewed the licensee's annual Radiological Environmental Surveillance Report for 1987, which was received April 19, 1988. The inspector had no questions regarding this report. The report was considered to be adequate.

No violations or deviations were identified.

12. Non-Radiological Confirmatory Measurements (79701)

To help assess the capability of the chemistry staff to perform acceptable analyses, a series of non-radiological chemistry samples were submitted to the licensee during a previous inspection (50-321/88-13 and 50-366/88-13). These "unknowns" were prepared for the NRC by Brookhaven National Laboratory. The licensee diluted the samples, as directed by the inspector, to bring the concentrations to within the ranges normally observed in plant aqueous systems. The results are presented in Attachment 1. The methodology for determining agreement between the licensee and NRC values is discussed in Attachment 2.

All samples were in agreement except for one sodium sample and all silica samples.

Although sodium sample 87J was in disagreement, it was within 8% of the NRC value.

Silica samples 87S, 87T and 87U were all biased low 23%, 13% and 17% respectively.

No violations or deviations were identified

13. Action On Previous Inspection Item (92701)

(Open) 50-321,366/87-11-02, Evaluate Adequacy of 30 Minute Time Lag Between Painting Operations and Charcoal Filter Testing. Licensee representatives stated that their evaluation was expected to be completed in mid July 1988. This item remained open.

(Closed) 50-321,366/87-25-01, Provide tritium analysis results of water samples to NRC. The licensee transmitted the result to NRC by letter of January 20, 1988. This matter is considered closed.

(Open) 50-321,366/87-25-02, Develop and implement procedure for verification of computer software. A licensee representative stated that licensee action was scheduled for completion approximately August 8, 1988.

(Closed) 50-321, 366/88-13-01 (IFI) Compare analytical results of chemistry samples. A licensee representative provided the inspector with the results of chemical analyses of "blind" chemistry samples. See Paragraph 12 of this report for details. This matter is considered closed.

14. Exit Interview

The inspection scope and results were summarized on July 1, 1988, with those persons indicated in Paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

Four internal audits conducted during 1987, identified several concerns within the scope of this inspection and adequate corrective measures were taken. Semi-Annual radioactive effluent release reports for 1987, were reviewed and determined to be adequate. A review of records of liquid and gaseous release permits and corresponding samples analyses, reactor coolant chemistry analysis logs, equipment test records, training, meteorology, and count-room cross-check records disclosed no discrepancies. The Environmental Report for 1987, and the Plant Annual Operating Report for 1987, were also reviewed.

15. Acronyms and Initialisms

| | |
|---|---|
| QA | Quality Assurance |
| RWCU | Reactor Water Clean-Up |
| ALARA | As Low As Reasonably Achievable |
| PASS | Post Accident Sampling System |
| SGTS | Standby Gas Treatment System |
| HEPA | High Efficiency Particulate Air |
| DOP | Di-octyl-phthalate |
| NBS | National Bureau of Standards |
| LLD | Lower Limit of Detection |
| ODCM | Offsite Dose Calculation Manual |
| SFSP | Spent Fuel Storage Pool |
| CFR | Code of Federal Regulations |

ATTACHMENT 1

NONRADIOLOGICAL INTERLABORATORY
TEST RESULTS
PLANT HATCH

| ANALYTE | ANALYSIS METHOD | DILUTION 1:X | NRC RESULTS Y ± s.d. (n) | LICENSEE RESULTS X ± s.d. (n) | RATIO (X/Y) Z | COMPARISON ± 2 s.d. |
|---|---|---|---|---|---|---|
| Chloride | | | | | | |
| 87A | IC | 1000 | 18.5 ± 0.1 (7) | 17.9 ± 0.6 (2)([1]) | 0.968 | Agreement |
| 87B | IC | 1000 | 37.3 ± 0.3 (7) | 39.1 ± 5.1 (3) | 1.048 | Agreement |
| 87C | IC | 2000 | 76.5 ± 1.2 (8) | 68.7 ± 12.6 (3) | 0.898 | Agreement |
| Sulfate | | | | | | |
| 87A | IC | 1000 | 19.5 ± 1.4 (7) | 19.9 ± 0.1 (2)([1]) | 1.021 | Agreement |
| 87B | IC | 1000 | 38.3 ± 2.7 (7) | 41.5 ± 4.8 (3) | 1.084 | Agreement |
| 87C | IC | 2000 | 78.0 ± 2.3 (9) | 87.2 ± 6.8 (3) | 1.118 | Agreement |
| Iron | | | | | | |
| 87G | AA (Flame) | 250 | 20.3 ± 0.6 (7) | 21.8 ± 2.2 (3) | 1.074 | Agreement |
| 87H | AA (Flame) | 1000 | 41.7 ± 0.7 (7) | 41.5 ± 1.4 (3) | 0.995 | Agreement |
| 87I | AA (Flame) | 1000 | 60.5 ± 2.5 (7) | 60.6 ± 2.8 (3) | 1.002 | Agreement |
| Copper | | | | | | |
| 87G | AA (Flame) | 1000 | 20.0 ± 0.3 (7) | 21.3 ± 1.5 (3) | 1.065 | Agreement |
| 87H | AA (Flame) | 1000 | 40.3 ± 1.5 (7) | 41.2 ± 1.3 (3) | 1.022 | Agreement |
| 87I | AA (Flame) | 1000 | 60.0 ± 1.5 (7) | 61.7 ± 1.4 (3) | 1.028 | Agreement |
| Nickel | | | | | | |
| 87G | AA (Flame) | 1000 | 20.3 ± 0.6 (7) | 21.5 ± 1.5 (3) | 1.059 | Agreement |
| 87H | AA (Flame) | 1000 | 41.7 ± 0.7 (7) | 41.3 ± 0.8 (3) | 0.990 | Agreement |
| 87I | AA (Flame) | 1000 | 60.5 ± 2.5 (7) | 61.0 ± 2.9 (6) | 1.008 | Agreement |
| Chromium | | | | | | |
| 87G | AA (Flame) | 1000 | 19.8 ± 0.5 (7) | 20.9 ± 1.0 (3) | 1.056 | Agreement |
| 87H | AA (Flame) | 1000 | 38.5 ± 0.5 (7) | 40.8 ± 2.0 (3) | 1.060 | Agreement |
| 87I | AA (Flame) | 1000 | 58.0 ± 1 (7) | 60.8 ± 3.7 (3) | 1.048 | Agreement |
| Sodium | | | | | | |
| 87J | IC | 2000 | 6.05 ± 0.7 (7) | 5.57 ± 0.7 (3) | 0.921 | Disagreement |
| 87K | IC | 2000 | 10.6 ± 0.6 (6) | 10.4 ± 1.0 (3) | 0.981 | Agreement |
| 87L | IC | 30µ 1 to 250ml | 15.8 ± 0.9 (6) | 15.6 ± 0.2 (3) | 0.987 | Agreement |
| Silica | | | | | | |
| 87S | vis spec | 1000 | 52.8 ± 2.8 (7) | 40.6 ± 2.1 (3) | 0.769 | Disagreement |
| 87T | vis spec | 1000 | 104 ± 4 (7) | 90.1 ± 3.8 (3) | 0.866 | Disagreement |
| 87U | vis spec | 1000 | 157 ± 2 (7) | 131 ± 5 (3) | 0.834 | Disagreement |

[1] Licensee analyzed unknown only two times because of insufficient sample amount. For comparison purposes, the licensee's value nearest the NRC's value was used as the third value in the comparison calculation.

## ATTACHMENT 2

## CRITERIA FOR COMPARING ANALYTICAL MEASUREMENTS

This attachment provides criteria for comparing results of the capability tests. The acceptance limits are based on the uncertainty (standard deviation) of the ratio for the licensee's mean value (X) to the NRC mean (Y), where

(1) $Z = X/Y$ is the ratio, and

(2) $S_z$ is the uncertainty of the ratio determined from the propagation of the uncertainties of licensee's mean value, $S_x$ and of the NRC's mean value, $S_y$.[1] Thus,

$$\frac{Sz^2}{Z^2} = \frac{Sx^2}{X^2} + \frac{Sy^2}{Y^2}$$

$$S_z = Z \left( \frac{Sx^2}{X^2} + \frac{Sy^2}{Y^2} \right)^{\frac{1}{2}}$$

The results are considered to be in agreement when the bias in the ratio (absolute value of difference between unity and the ratio) is less than or equal to twice the uncertainty in the ratio, i.e.,:

$$| 1-Z | \leq 2 S_z$$

[1] National Council on Radiation Protection and Measurements, A Handbook of Radioactivity Measurement Procedures, NCRP Report No. 58, Second Edition, 1985, Pages 322-326 (see Page 324).