VERMONT YANKEE NUCLEAR POWER CORPORATION

PROCESS CONTROL PROGRAM

REV 5

4/23/96

Submitted J.J. Geyster for MPD
Radiation Protection Manager

Approved M. M. Houle Mtg. 96-047
PORC

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APPENDIX A

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT
Supplemental Information
for 1996

Facility: Vermont Yankee Nuclear Power Station

Licensee: Vermont Yankee Nuclear Power Corporation

1A. TECHNICAL SPECIFICATION LIMITS - DOSE AND DOSE RATE

Technical Specification and Category		Limit
a. Noble Gases		
3.8.E.1	Total body dose rate	500 mrem/yr
3.8.E.1	Skin dose rate	3000 mrem/yr
3.8.F.1	Gamma air dose	5 mrad in a quarter
3.8.F.1	Gamma air dose	10 mrad in a year
3.8.F.1	Beta air dose	10 mrad in a quarter
3.8.F.1	Beta air dose	20 mrad in a year
b. Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form With Half-Lives Greater Than 8 Days		
3.8.E.1	Organ dose rate	1500 mrem/yr
3.8.G.1	Organ dose	7.5 mrem in a quarter
3.8.G.1	Organ dose	15 mrem in a year
c. Liquids		
3.8.B.1	Total body dose	1.5 mrem in a quarter
3.8.B.1	Total body dose	3 mrem in a year
3.8.B.1	Organ dose	5 mrem in a quarter
3.8.B.1	Organ dose	10 mrem in a year

APPENDIX A
(Continued)

EFFLUENT AND WASTE DISPOSAL ANNUAL REPORT

2A. TECHNICAL SPECIFICATION LIMITS - CONCENTRATION

Technical Specification and Category	Limit
a. Noble Gases	No MPC Limits (No ECL Limits)
b. Iodine-131, Iodine-133, Tritium and Radionuclides in Particulate Form With Half-Lives Greater Than 8 Days	No MPC Limits (No ECL Limits)
c Liquids	
3.8.A.1 Total fraction of MPC (ECL) excluding noble gases (10CFR20, Appendix B, Table II, Column 2):	$\leq$1.0
3.8.A.1 Total noble gas concentration:	$\leq$2E-04 µCi/cc

3. AVERAGE ENERGY

Provided below are the average energy ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases, if applicable.

a. Average gamma energy: Not Applicable

b. Average beta energy: Not Applicable

4. MEASUREMENTS AND APPROXIMATIONS OF TOTAL RADIOACTIVITY

Provided below are the methods used to measure or approximate the total radioactivity in effluents and the methods used to determine radionuclide composition.

a. Fission and Activation Gases

Continuous stack monitors monitor the gross Noble Gas radioactivity released from the plant stack. Because release rates are normally below the detection limit of these monitors, periodic grab samples are taken and analyzed for the gaseous isotopes present. These are used to calculate the individual isotopic releases indicated in Table 1B and the totals of Table 1A. The error involved in these steps may be approximately ±23 percent.

b. Iodines

Continuous isokinetic samples are drawn from the plant stack through a particulate filter and charcoal cartridge. The filters and cartridges are normally removed weekly and are analyzed for Iodine-131, 132, 133, 134, and 135. The error involved in these steps may be approximately ±18 percent.

c. Particulates

The particulate filters described in b. above are also counted for particulate radioactivity. The error involved in this sample is also approximately ±18 percent.

d. Tritium

Grab samples from the plant stack are taken monthly through a cold trap collection device and analyzed for tritium. The error involved in this sample is approximately ±15 percent.

e. Waste Oil

Prior to issuing the permit to burn a drum of radioactively contaminated waste oil, one liter of the oil is analyzed by gamma spectroscopy to determine concentrations of radionuclides that meet or exceed the LLD for all of the liquid phase radionuclides listed in Technical Specification Table 4.8.1. Samples that have a visible water layer are not analyzed. The water must first be removed from the drum of oil and resampled.

Monthly, samples from drums that were issued burn permits are sent to the E-Lab for compositing and analysis. The E-Lab analyzes for tritium, alpha, Fe-55, Sr-89, and Sr-90 on the composite sample.

The waste oil samples are liquid effluents that end up as a gaseous ground level release.

f. Liquid Effluents

Radioactive liquid effluents released from the facility are continuously monitored. Measurements are also made on a representative sample of each batch of radioactive liquid effluents released. For each batch, station records are retained of the total activity (mCi) released, concentration (µCi/ml) of gross radioactivity, volume (liters), and approximate total quantity of water (liters) used to dilute the liquid effluent prior to release to the Connecticut River.

Each batch of radioactive liquid effluent releases is analyzed for gross gamma and gamma isotopic radioactivity. A monthly proportional composite sample, comprising an aliquot of each batch released during a month, is analyzed for tritium and gross alpha radioactivity. A quarterly proportional composite sample, comprising an aliquot of each batch released during a quarter, is analyzed for Sr-89, Sr-90, and Fe-55.

5. BATCH RELEASES

a. Liquid

There were no routine liquid batch releases during the reporting period.

b. Gaseous

There were no routine gaseous batch releases during the reporting period.

6. ABNORMAL RELEASES

a. Liquid

There were no nonroutine liquid releases during the reporting period.

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b. Gaseous

There were no nonroutine gaseous releases during the reporting period.

APPENDIX B

LIQUID HOLDUP TANKS

Requirement: Technical Specification 3.8.D.1 limits the quantity of radioactive material contained in any outside tank. With the quantity of radioactive material in any outside tank exceeding the limits of Technical Specification 3.8.D.1, a description of the events leading to this condition is required in the next Annual Effluent Release Report per Technical Specification 6.7.C.1.

Response: The limits of Technical Specification 3.8.D.1 were not exceeded during this reporting period.

APPENDIX C

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Requirement: Radioactive liquid effluent monitoring instrumentation channels are required to be operable in accordance with Technical Specification Table 3.9.1. If an inoperable radioactive liquid effluent monitoring instrument is not returned to operable status prior to a release pursuant to Note 4 of Table 3.9.1, an explanation in the next Annual Effluent Release Report of the reason(s) for delay in correcting the inoperability are required per Technical Specification 6.7.C.1.

Response: Since the requirements of Technical Specification Table 3.9.1 governing the operability of radioactive liquid effluent monitoring instrumentation were met for this reporting period, no response is required.

APPENDIX D

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

<u>Requirement</u>: Radioactive gaseous effluent monitoring instrumentation channels are required to be operable in accordance with Technical Specification Table 3.9.2. If inoperable gaseous effluent monitoring instrumentation is not returned to operable status within 30 days pursuant to Note 5 of Table 3.9.2, an explanation in the next Annual Effluent Release Report of the reason(s) for the delay in correcting the inoperability is required per Technical Specification 6.7.C.1.

<u>Response</u>: Since the requirements of Technical Specification Table 3.9.2 governing the operability of radioactive gaseous effluent monitoring instrumentation were met for this reporting period, no response is required

APPENDIX E

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Requirement: The radiological environmental monitoring program is conducted in accordance with Technical Specification 3.9.C. With milk samples no longer available from one or more of the sample locations required by Technical Specification Table 3.9.3, Technical Specification 6.7.C.1 requires the following to be included in the next Annual Effluent Release Report: (1) identify the cause(s) of the sample(s) no longer being available, (2) identify the new location(s) for obtaining available replacement samples and (3) include revised ODCM figure(s) and table(s) reflecting the new location(s).

Response: The Back Tracks Farm in Vernon, Vermont went out of business on April 9, 1996. As a consequence, Table 4.1, Parts 3.a and 3.c, as well as Figure 4-2 of the ODCM, have been revised to remove this milk and silage sampling location from the Radiological Environmental Monitoring Program (REMP). No new replacement stations need to be added to the program since the required number of highest-dose-potential farms, as listed in the 1995 Land Use Census, are already part of the REMP. The revised ODCM figure and table are included in the attachment to Appendix H.

APPENDIX F

LAND USE CENSUS

Requirement: A land use census is conducted in accordance with Technical Specification 3.9.D. With a land use census identifying a location(s) which yields at least a 20 percent greater dose or dose commitment than the values currently being calculated in Technical Specification 4.8.G.1, Technical Specification 6.7.C.1 requires the identification of the new location(s) in the next Annual Effluent Release Report.

Response: The Land Use Census was completed in the third quarter of 1996. No locations yielded a 20 percent greater dose or dose commitment than the values currently being calculated in Technical Specification 4.8.G.1.

APPENDIX G

PROCESS CONTROL PROGRAM

Requirement: Technical Specification 6.12.A.1 requires that licensee initiated changes to the Process Control Program (PCP) be submitted to the Commission in the Annual Radioactive Effluent Release Report for the period in which the change(s) was made.

Response: The following changes were made to the Process Control Program (PCP) and issued as Revision 5 during this reporting period.

Title Page:	Changed approval by Operations Manager to VP/MOO as stated in Tech Specs.
Section 1:	Added reference to TS 4.8.N; added statement to review and approve vendor's PCP program by PORC and VP/MOO.
Section 2:	Added statement to air dry filters 24 hours or as determined by RWS. Also added method to dispose of filters above dose limitations per 49 CFR.
Section 3:	Added reference to OP-2153. Remove reference to remote dewatering which is no longer used.
Section 4:	Removed reference concerning the use of a vacuum pump to dewater the liners. This method is no longer used.
Section 5:	Added reference to OP-2512.
Section 6:	None.
Section 7:	Added reference to MSDS via AP-0620.
Section 8:	Same as Section 7.
Section 9:	Same as Section 7.
Section 10:	None.

APPENDIX G
(Continued)

PROCESS CONTROL PROGRAM

Section 11: Added reference to the use of an approved outside lab.

Section 12: Added this new section - Mixed Waste.

Page 4 was revised to incorporate procedure title and revision changes, OP-2511 and OP-2512.

This revision does not affect Technical Specifications and does not affect any system or processes described in the FSAR.

The revised Process Control Program is attached.

VERMONT YANKEE NUCLEAR POWER CORPORATION

PROCESS CONTROL PROGRAM

REV 5

4/23/96

Submitted ______ for MPD
Radiation Protection Manager

Approved M. M. Houle Mtg. 96-047
PORC

Approved ______
Plant Manager

Approved ______
VP, MOO

VERMONT YANKEE NUCLEAR POWER CORPORATION

PROCESS CONTROL PROGRAM

Introduction:

The Vermont Yankee Nuclear Power Corporation Process Control Program (PCP) describes the administrative and technical controls on the radioactive waste systems which provide assurance that Vermont Yankee meets federal shipping and burial site requirements.

The PCP complies with Technical Specification 6.12 by describing process parameters, controls, tests, sampling and analysis to ensure compliance with 10 CFR 20, 10 CFR 71, and 10 CFR 61 Energy, 49 CFR 172-173 Transportation, state, and burial site regulation requirements.

1.0 Solidification

Vermont Yankee Nuclear Power Corporation does not routinely solidify liquid waste. If the use of solidification to dispose of any liquid waste is required, it will be done by an outside vendor under the vendor's PCP. The vendor PCP will be reviewed and approved by the Plant Health Physicist, the Radiation Protection Manager, PORC and VP, MOO prior to implementation. This review is to identify that there is sufficient supporting documentation of the vendor's PCP to give assurance that the final product will meet all requirements for transport and burial, and that sufficient procedural controls exist to assure safe operations. [TS 4.8.N]

2.0 Cartridge Filter Elements

Low activity cartridge filter elements will be air dried (~24hr or as determined by the Radwaste Supervisor) and handled as dry active waste. Filters determined to be above the dose limitations per 49CFR, will be placed in casks and shipped for disposal.

3.0 Resins

Normal operations produce radioactive waste in the form of depleted resins. These resins are processed in the burial container using a rapid dewatering system (RDS-1000) manufactured by Chem-Nuclear Systems, Inc. [OP 2153]

The system has been tested, by Chem-Nuclear, for certification in meeting the Barnwell Site Criteria and disposal requirement for free standing liquid. These tests are described in Chem-Nuclear's Topical Report on the RDS-1000 Radioactive Waste Dewatering System. In addition, to comply with the statement, "Any liquids present in waste packages shall be non-corrosive with respect to the container," Vermont Yankee tested the pH of various resin mixtures used by the plant in solution with water. The range was found to be 4.2 - 8.4. A solution is not considered corrosive if the pH is greater than 4.0.

A resin sample is taken from each liner prior to shipment. The sample is counted to determine the activity and waste classification. Class A resins that exceed 1.0 μCi/cc of isotopes with greater than 5 year half-lives and all Class B and C resins will be disposed of in an approved High Integrity Container (HIC).

Vendor supplied or temporary methods of processing resins may be used in lieu of the above process provided that the vendor or temporary process meets the requirements of quality described above and does not conflict with accepted burial criteria or safety requirements. Such methods will be reviewed and approved by the Plant Health Physicist and the Radiation Protection Manager prior to implementation.

4.0 Filter Liners

During refueling outages and normal operation, liquid radwaste processing may require use of a decanting filter on the condensate phase separators. A floating suction is used to decant the water and resin into a filter liner. Filtered water is pumped from the liner. The liner is dewatered in accordance with OP 2511 (MOOID9409-03) such that the burial site criterion for free-standing water is met. A resin sample is taken from the liner and analyzed to determine the activity and waste classification.

5.0 Dry Active Waste (DAW)

DAW is compacted, as practical, or shipped to a vendor that sorts the material for processing or recycling. All DAW is examined before being compacted or shipped. Any liquids or items found that would compromise the integrity of the package are removed and separated as specified by procedure. [OP 2512] Containers used for DAW shipments meet the criteria of 49 CFR 173.425a. or b. "No leakage of radioactive material," as specified in 49 CFR 173.425.b.1 will be met provided that no radioactive materials in quantities equal to or exceeding those specified in 49 CFR 173.443 are detected on the external surfaces of the package at any time during shipment.

6.0 Chelating Agents

In order to comply with 10 CFR 20 Appendix F, chelating agents are controlled by the plant chemistry department using procedure AP 0620.

7.0 Explosive Waste

No waste capable of detonation or of explosive decomposition or reaction will be disposed as per 10 CFR 61.56(a)(4). Refer to MSDS via AP 0620.

8.0 Toxic Waste

No waste capable of generating toxic gases, vapors, or fumes will be disposed as per 10 CFR 61.56(a)(5). Refer to MSDS via AP 0620.

9.0 Pyrophoric Waste

No waste that is pyrophoric will be disposed as per 10 CFR 61.56(a)(6). Refer to MSDS via AP 0620.

10.0 High Integrity Containers (HICs)

Vermont Yankee Nuclear Power Corporation has contracted with various suppliers of approved HICs. South Carolina has approved PCPs for HICs used by Vermont Yankee. Any HIC Vermont Yankee may choose to use at some future time, will meet all applicable requirements.

11.0 Waste Class Determination

Along with an approved outside laboratory, Vermont Yankee periodically performs laboratory analysis on all waste streams to determine the activity of radionuclides listed in Tables 1 and 2 of 10 CFR 61. Correlation analysis verifies that the relative concentration of each radionuclide, with respect to the overall activity in a given Vermont Yankee waste stream, remains constant over time. A set of scaling factors is determined which allows the activity of 10 CFR 61 radionuclides to be estimated using the results of gamma spectrometric analysis or direct gamma dose rate measurements.

For resin wastes, analysis is performed on samples of each source of resin comprising the contents of a burial container. Scaling factors are applied to the activity of radionuclides identified by gamma spectrometry analysis to determine the activity of those radionuclides which are not detected in the gamma spectrum.

For DAW, dose rate-to-curie conversion calculations are performed to determine the total activity present in a container. Scaling factors are applied to the container's total curie content to determine the activity of individual radionuclides.

Specific procedures for determining 10 CFR 61 scaling factors are contained in OP 2527, "Sampling and Analysis for Radwaste Classification." Once the activity of each radionuclide in a burial container is estimated, the waste classification is derived using methods required by 10 CFR 61. Specific procedures for waste class determination are contained in AP 0504, "Shipment of Radioactive Material."

12.0 Mixed Waste

No mixed waste will be disposed as per 10 CFR 61.56(a)(8) unless properly treated. Refer to MSDS via AP 0620.

PROCEDURES WHICH IMPLEMENT THE PCP

1.	AP 0504	Shipment of Radioactive Materials
2.	OP 2511	Radwaste Cask/Liner Handling
3.	OP 2527	Sampling and Analysis for Radwaste Classification
4.	OP 2151	Liquid Radwaste
5.	OP 2153	Solid Radwaste
6.	AP 0620	Chemical Material Use
7.	OP 2512	Radwaste Drum, Box and Sealand Handling

APPENDIX H

OFF-SITE DOSE CALCULATION MANUAL

Requirement: Technical Specification 6.13.A.1 requires that licensee initiated changes to the Off-Site Dose Calculation Manual (ODCM) be submitted to the Commission in the Annual Radioactive Effluent Release Report for the period in which the change(s) was made effective.

Response: Revision 20 to the Off-Site Dose Calculation Manual was made during this reporting period.

The major changes included in Revision 20 to the ODCM are:

(1) Page 3-51: Direct off-site dose due to N-16 decay from the turbine is calculated using Equation 3-27 in the ODCM. One variable in this equation is the gross electric output during periods of interest. In 1995, new turbine rotors were installed which had the effect of increasing the electric output of the plant due to higher turbine efficiency. However, the actual mechanism upon which the dose from N-16 is derived, involved the carryover of N-16 in the steam flow from the reactor through the turbine, and not directly the electric output of the plant. The measurements that related N-16 dose with power level used gross electric output as a readily available parameter to represent the relative amount of steam that passed through the turbine. With the increase in efficiency due to the new turbine rotors, the correlation between steam flow and electric output has changed slightly (i.e., there is more gross electric output for the same steam flow, and therefore the same N-16 carryover as before). The original turbine generator heat balance (FSAR Figure 1.6-2) indicated a gross heat rate of 10,123 Btu/kWhr. The revised calculated heat balance (General Electric Co. Drawing 510HB362, Rev. 2) shows a gross heat rate of 9891 Btu/kWhr, or approximately 2.3% increase in efficiency. As a consequence, a conservative 2% reduction in the N-16 dose conversion factor of ODCM Equation 3-27 has been made to adjust the expected dose as a function of new gross electric output of the turbine.

(2) Page 3-5: The old turbine rotors (and casings), as noted above, were placed in storage sheds near the west site boundary. Surveys of the rotors and casings indicate low level surface contamination (principally Co-60). Dose rate projections from these components based on the surveys taken at the storage location indicate that the annual impact on the maximum west site boundary from direct dose is about 0.2 mrem. Since the turbine hall and other fixed sources already account for direct radiation doses on the order of about 15 mrem/year (reported to the nearest tenth of a mrem), dose contributions of less than 0.1 mrem/year are not considered significant with respect to their impact on the total dose. A commitment is added to the ODCM to include in the annual estimate of total direct dose from all sources the projected site boundary dose from the turbine rotor storage sheds as long as the rotors are in storage and the projected dose from them is greater than 0.1 mrem/year.

(3) Table 1.1-12: Isotopic analysis of recent gas samples indicates, that at times, Co-57 is detected in effluent releases to the environment. This radionuclide was not included in the original listing of dose factors for Method I dose calculations since it was not included in the data library listings of Regulatory Guide 1.109 which formed the bases of all ODCM dose factors. In response to the identification of Co-57 in gas effluents, new dose factors have been developed using the same modeling approach and data bases as used by the NRC in the derivation of the original set of Regulatory Guide 1.109 dose factors. The resulting dose factors are, therefore, consistent with the methods already accepted and used in Regulatory Guide 1.109 and the ODCM. In addition, new dose factors for two other radionuclides (Se-75 and Sn-113), that were not included in the regulatory guide listing, have been developed in the same way as for Co-57. This is because other plants have reported detecting these radionuclides in effluents. Table 1.1-12 dose factors have been recalculated with the expanded library to add these three radionuclides to the ODCM listing.

It should also be noted that the recalculation of I-133 resulted in a slight difference in the last decimal place of the original dose factor listed in the ODCM. This is believed to be the result of a slight change in the last digit round off due to the use of an updated half-life value for this radionuclide. No significant differences in off-site doses would result from this change, but it should be reflected in plant procedures to ensure consistency with the ODCM.

(4) Page vi, Appendix E: By letters, dated November 18, 1991 and July 10, 1992 Vermont Yankee submitted to the NRC a request for approval of proposed procedures for in-place disposal of slightly contaminated radioactive soil under the Chem Lab floor which had resulted from a break in a chem sink drain line. The NRC responded to this request with their approval (letter NVY 96-48, dated March 7, 1996) documented in a Safety Evaluation. The NRC acceptance of our plans to dispose of the soil in-place required that their SE be incorporated into the ODCM as an Appendix. A new Appendix E containing the NRC SE has been added to the ODCM to complete this requirement (LAI No. 1130B).

(5) Pages 4-2, 4-2a, and 4-5: On April 9, 1996, it was identified that the Back Tracks Farm in Vernon, Vermont (location TM-10 and TC-10) had gone out of business. It has been added to the Radiological Environmental Monitoring Program (REMP) in the fall of 1995 as a result of the 1995 Land Use Census ranking it among the highest potential receptor locations that warranted inclusion in the monitoring program. As a consequence, Table 4.1, Parts 3.a and 3.c, as well as Figure 4-2, have been revised to remove this milk and silage sampling location from the REMP. No new replacement stations need to be added to the program since the required number of highest-dose-potential farms, as listed in the 1995 Land Use Census, are already part of the REMP.

(6) Page 5-17: Current plant calibration procedures use Xe-133 as the reference radionuclide to determine counting

efficiencies of the Stack Gas I and II, and AOG monitors. This is based on the combination of its expected presence in the off-gas, and a relatively long half-life that makes it practical to detect before decay eliminates it. Xe-133 is a commonly used gas in the industry for monitor calibrations. During normal plant operations where off-gas is processed through the AOG, short-lived gases are decayed away before being seen by the monitors, leaving only long-lived Xe-133 and Kr-85. The use of Xe-133 instead of Kr-85, or a mix of the two, is based on the fact that the expected response of the detectors is conservative for Xe-133 in comparison to Kr-85. If short-lived noble gases were present in significant amounts due to some off-normal condition, the use of Xe-133 as a reference gas for counting efficiency would still lead to a conservative monitor response relative to Xe-133 since short-lived gas mixes tend to have higher response factors (i.e., high energy noble gases will cause the monitor to respond as if a higher concentration of radioactivity were being discharged than actually is). The new wording also clarifies that grab samples and lab analysis of gas releases can be used to determine the release rate of noble gases from the stack or AOG at any time. This ensures operational flexibility if monitor readings are suspected of over responding to actual release conditions. This is considered an administrative change to simplify plant procedures.

(7) Page 6-10: Figure 6-2 illustrates the basic operational layout of the Advanced Off-Gas (AOG) System. Based on the results of an AOG System review, Figure 6-2 has been corrected to indicate that the AOG bypass line that provides a discharge path around the moisture removal/dryer subsystem and charcoal adsorber subsystem feeds back into the main discharge line downstream of the AOG noble gas monitors (RAN-OG-3127 and RAN-OG-3128) and before the final 60% section of the original delay pipe. This had been incorrectly shown that the bypass line returned gas flow upstream of these two monitors. This is considered an administrative change to reflect "as built" conditions.

OFF-SITE DOSE CALCULATION MANUAL

(8) Page 4-1: In 1995, the EPA announced that they would not be able to supply a full range of environmental control samples in 1996 for use in laboratory intercomparison studies program in which all nuclear power plants are required to participate. Vermont Technical Specification 3/4.9.E requires that the approved NRC Intercomparison Program used by the laboratory performing environmental analyses be identified in the ODCM. Section 4.1 of the ODCM has been updated to indicate the various NIST traceable quality assurance programs in which the YAEL participates. A recent NRC Region I inspection of the YAEL QA Programs concluded that the intercomparison studies in which the YAEL participates satisfy NRC requirements to meet the Technical Specification commitments.

(9) Pages 3-14 and 3-20: A clarification has been included in the definition of stack release quantities that provides for direct isotopic measurement (grab samples with laboratory analysis) of the radionuclide mixes in the gas stream being discharged from the stack, and the alternate method of inferring the radionuclide mixes at the stack by using SJAE measurements. These two approaches are part of current plant procedures, and provides operational flexibility to estimate the continuous radionuclide release rate from the plant stack. This is considered an administrative change to reflect existing practices.

(10) Pages 5-13, 5-14, 5-15, and Appendix A: With the routing of the turbine hall exhaust air to the plant stack, and the replacement of the stack gas monitors (I and II) with new detectors, both effluent flow rate and monitor response factors have changed slightly from values originally used in example calculations used to illustrate the use of various dose and dose rate applications. This revision updates these illustrations to provide plant staff with examples that reflect information more relevant to the current plant configuration. These changes are considered administrative in nature in that no change is made in the methodology used to calculate doses.

OFF-SITE DOSE CALCULATION MANUAL

The above changes will not reduce the accuracy or reliability of the dose calculations or setpoint determinations previously approved for use in the ODCM. This conclusion is based on the nature of the changes that: (a) adjust existing methodology for calculating direct dose by factoring changes in turbine efficiency to maintain the same level of N-16 impact as a function of electric power as previously used, (b) include additional fixed radiation sources in estimating a better total site boundary dose, (c) use the existing dose methodology to expand the listing of radionuclides that can be considered in calculating Method I dose impacts, (d) expand the description of gas monitor calibrations to reference Xe-133 as the standard reference gas in keeping with common industry practices, (e) clarify existing plant procedures for direct measurement of effluent gas mixes in addition to estimates derived from SJAE measurement, and (f) describe administrative changes that update system flow diagrams to existing plant configurations, add new documentation for the approval to dispose of slightly contaminated soil, replace REMP sampling locations due to sample unavailability, describe the currently approved intercomparison QA Program at the YAEL, and show examples of the type of calculations that can be addressed by the ODCM methods.

The revised pages from Revision 20 to the ODCM are attached.