October 9, 1984

Docket No. 50-271

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

The Commission has issued the enclosed Amendment No. 83 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your January 23, 1984 application.

The amendment authorizes changes to the Technical Specifications (1) to implement the requirements of Appendix I of 10 CFR Part 50, (2) to establish new limiting conditions for operation for the quarterly and annual average release rates, and (3) to revise environmental monitoring programs to assure conformance with the Commission's regulations.

As part of your correspondence on this subject, you submitted copies of the Offsite Dose Calculation Manual. We have reviewed the Offsite Dose Calculation Manual and have found it acceptable. To facilitate implementation of the Vermont Yankee Radiological Effluent Technical Specifications, your omission of monitors for turbine building effluents is accepted on an interim basis until a study by the NRC of monitoring alternatives for the turbine building has been completed.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Vernon L. Rooney, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 83 to License No. DPR-28
2. Safety Evaluation

see "Tech Specs"

cc w/enclosures:
See next page

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Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation
Vermont Yankee Nuclear Power Station

cc:

Mr. W. F. Conway
President & Chief Executive Officer
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Donald Hunter, Vice President
Vermont Yankee Nuclear Power Corp.
1671 Worcester Road
Framingham, Massachusetts 01701

New England Coalition on
Nuclear Pollution
Hill and Dale Farm
R. D. 2, Box 223
Putney, Vermont 05346

Mr. Walter Zaluzny
Chairman, Board of Selectman
Post Office Box 116
Vernon, Vermont 05345

J. P. Pelletier, Plant Manager
Vermont Yankee Nuclear Power Corp.
Post Office Box 157
Vernon, Vermont 05354

Raymond N. McCandless
Vermont Division of Occupational
& Radiological Health
Administration Building
10 Baldwin Street
Montpelier, Vermont 05602

Honorable John J. Easton
Attorney General
State of Vermont
109 State Street
Montpelier, Vermont 05602

John A. Ritscher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

W. P. Murphy, Vice President &
Manager of Operations
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

U. S. Environmental Protection
Agency
Region I Office
Regional Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Vermont Yankee Decommissioning
Alliance
43 State Street
Montpelier, Vermont 05602-2964

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 176
Vernon, Vermont 05354

Vermont Public Interest
Research Group, Inc.
43 State Street
Montpelier, Vermont 05602

Thomas A. Murley
Regional Administrator
Region I Office
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Richard Saudek, Commissioner
Vermont Department of Public Service
120 State Street
Montpelier, Vermont 05602

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated January 23, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-28 is hereby amended to read as follows:

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 83, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of April 1, 1985.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 9, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise the Technical Specifications as follows:

Remove	Insert
iii	iii
iv	iv
-	v
2	2
-	2a
-	4a
-	4b
46	46
58	58
65	65
147 through 160	147 through 160
-	160a through 160i
161	161
-	161a
162 through 172	162 through 172
-	172a through 172l
180c	180c
180h	180h
188	188
197	197
198	198
198a	198a
200	200
201	201
206	206
207a	207a
208 through 215	208 through 222

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(continued)

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(continued)

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(continued)

G. <u>Instrument Functional Test</u> - An instrument functional test shall be:

1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

2. Bistable channels - the injection of a signal into the sensor to verify operability including alarm and/or trip functions.

H. <u>Logic System Functional Test</u> - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

I. <u>Minimum Critical Power Ratio</u> - The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the GEXL correlation to the actual assembly operating power (Reference NEDO-10958).

J. <u>Mode</u> - The reactor mode is that which is established by the mode-selector-switch.

K. <u>Operable</u> - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

L. <u>Operating</u> - Operating means that a system or component is performing its intended functions in its required manner.

M. <u>Operating Cycle</u> - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. <u>Primary Containment Integrity</u> - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Rev. 1, October 1977.

DD. Solidification - Solidification shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements. Suitable forms include dewatered resins and filter sludges.

EE. Member(s) of the Public - Members of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are casual visitors to the plant and persons who enter the site to service equipment or to make deliveries.

FF. Site Boundary - The site boundary is shown in Figure 2.2-5 in the FSAR.

GG. Radioactive Material - Any material or combination of materials which spontaneously emits ionizing radiation and in which the specific activity is greater than 0.002 microcuries/gram of material or any material in which the total estimated activity is greater than 5 microcuries is classified as a radioactive material.

HH. Contamination

1. Removable radioactive contamination shall be considered significant and unreleasable from the owner controlled area if the level, when averaged over 300 square centimeters, exceeds 220 dpm/ square centimeter for beta-gamma, and 22 dpm/square centimeter for alpha emitting radionuclides.

2. Fixed contamination shall be considered significant and unreleasable from the owner controlled area if the dose rate at any accessible surface exceeds 0.5 mrem/hour.

II. Off-Site Dose Calculation Manual (ODCM) - A manual containing the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduction of the environmental radiological monitoring program.

JJ. Process Control Program (PCP) - A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

KK. Gaseous Radwaste Treatment System - The Augmented Off-Gas System (AOG) is the gaseous radwaste treatment system which has been designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

LL. Ventilation Exhaust Treatment System - The Radwaste Building and AOG Building ventilation HEPA filters are ventilation exhaust treatment systems which have been designed and installed to reduce radioactive material in particulate form in gaseous effluents by passing ventilation air through HEPA filters for the purpose of removing radioactive particulates from the gaseous exhaust stream prior to release to the environment. Engineered safety feature atmospheric cleanup systems, such as the Standby Gas Treatment (SBGT) System, are not considered to be ventilation exhaust treatment system components.

MM. Vent/Purging - Vent/Purging is the controlled process of discharging air or gas from the primary containment to control temperature, pressure, humidity, concentration or other operating conditions.

TABLE 3.2.4

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Condition for Operation Are Not Met
1	Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD)	$\leq$ 2 minutes $\leq$ 30 minutes	Note 1
1	Trip System Logic	---	Note 1

Note 1 – At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

TABLE 4.2.4

MINIMUM TEST AND CALIBRATION FREQUENCIES

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Augmented Off-Gas			
Trip System Logic (AOG)	Every 6 months (Note 2)	Every 6 months (Note 3)	---

2.2 (continued)

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches the fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

3.8 LIMITING CONDITIONS FOR OPERATION

3.8 RADIOACTIVE EFFLUENTS

Applicability

Applies to the release of all radioactive effluents from the plant.

Objective

To assure that radioactive effluents are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the limits specified in 10CFR20.

Specification

A. Liquid Effluents: Concentration

1. The concentration of radioactive material in liquid effluents released from the site shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases and $2x10^{-4}$ uCi/ml total activity concentration for all dissolved or entrained noble gases.

4.8 SURVEILLANCE REQUIREMENTS

4.8 RADIOACTIVE EFFLUENTS

Applicability

Applies to the required surveillance of all radioactive effluents released from the plant.

Objective

To ascertain that all radioactive effluents released from the plant are kept "as low as is reasonably achievable" in accordance with 10CFR50, Appendix I and, in any event, are within the limits specified in 10CFR20.

Specification

A. Liquid Effluents: Concentration

1. Radioactive material in liquid waste shall be sampled and analyzed in accordance with requirements of Table 4.8.1. The results of the analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are limited to the values in Specification 3.8.A.1.

3.8 LIMITING CONDITIONS FOR OPERATION

2. With the concentration of radioactive material in liquid effluents released from the site exceeding the limits of Specification 3.8.A.1, immediately take action to decrease the release rate of radioactive materials and/or increase the dilution flow rate to restore the concentration to within the above limits.

B. Liquid Effluents: Dose

1. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site shall be limited to the following:

   a. During any calendar quarter:

      less than or equal to 1.5 mrem to the total body, and

      less than or equal to 5 mrem to any organ, and

   b. During any calendar year:

      less than or equal to 3 mrem to the total body, and

      less than or equal to 10 mrem to any organ.

4.8 SURVEILLANCE REQUIREMENTS

B. Liquid Effluents: Dose

1. Cumulative dose contributions shall be determined in accordance with the methods in the ODCM at least once per month if releases during the period have occurred.

3.8 LIMITING CONDITIONS FOR OPERATION

C. Liquid Radwaste Treatment

1. The liquid radwaste treatment system shall be used in its designed modes of operation to reduce the radioactive materials in the liquid waste prior to its discharge when the estimated doses due to the liquid effluent from the site, when averaged with all other liquid release over the last month, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.

D. Liquid Holdup Tanks

1. The quantity of radioactive material contained in any outside tank* shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

4.8 SURVEILLANCE REQUIREMENTS

C. Liquid Radwaste Treatment

1. See Specification 4.8.B.1.

D. Liquid Holdup Tanks

1. The quantity of radioactive material contained in each of the liquid holdup tanks* shall be determined to be within the limits of Specification 3.8.D.1 by analyzing a representative sample of the tank's content at least once per week or when radioactive materials are being added to the tank.

---

*NOTE: Tanks included in this Specification are only those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank's contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

3.8 LIMITING CONDITIONS FOR OPERATION

2. With the quantity of radioactive material in any outside tank* exceeding the limit of Specification 3.8.D.1, immediately take action to suspend all additions of radioactive material to the tank. Within 48 hours, reduce the tank contents to within the limit.

E. Gaseous Effluents: Dose Rate

1. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

   a. For noble gases; less than or equal to 500 mrem/yr to the total body and less than or equal to 3,000 mrem/yr to the skin, and

   b. For Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days; less than or equal to 1,500 mrem/yr to any organ.

2. With the dose rate(s) exceeding the above limits, immediately take action to decrease the release rate to within the limits of Specification 3.8.E.1.

4.8 SURVEILLANCE REQUIREMENTS

E. Gaseous Effluents: Dose Rate

1. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.8.E.1 in accordance with the methods in the ODCM.

2. The dose rate due to Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.8.E.1 in accordance with the methods in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.8.2.

3.8 LIMITING CONDITIONS FOR OPERATION

F. Gaseous Effluents: Dose from Noble Gases

1. The air dose due to noble gases released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

   a. During any calendar quarter:

      less than or equal to 5 mrad for gamma radiation, and

      less than or equal to 10 mrad for beta radiation, and

   b. During any calendar year:

      less than or equal to 10 mrad for gamma radiation, and

      less than or equal to 20 mrad for beta radiation.

4.8 SURVEILLANCE REQUIREMENTS

F. Gaseous Effluents: Dose from Noble Gases

1. Cumulative dose contributions for the total time period shall be determined in accordance with the methods in the ODCM at least once every month.

3.8 LIMITING CONDITIONS FOR OPERATION

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

1. The dose to a member of the public from Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the site boundary shall be limited to the following:

   a. During any calendar quarter: less than or equal to 7.5 mrem to any organ, and

   b. During any calendar year: less than or equal to 15 mrem to any organ.

H. Gaseous Radwaste Treatment

1. The Augmented Off-Gas System (AOG) shall be used in its designed mode of operation to reduce noble gases in gaseous waste prior to their discharge whenever the main condenser steam jet air ejector (SJAE) is in operation.

4.8 SURVEILLANCE REQUIREMENTS

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

1. Cumulative dose contributions for the total time period shall be determined in accordance with the methods in the ODCM at least once every month.

H. Gaseous Radwaste Treatment

1. The readings of the relevant instrument shall be checked every 12 hours when the main condenser SJAE is in use to ensure that the AOG is functioning.

3.8 LIMITING CONDITIONS FOR OPERATION

I. Ventilation Exhaust Treatment

1. The AOG and Radwaste Building Ventilation Filter (HEPA) Systems shall be used to reduce particulate materials in gaseous waste prior to their discharge from those buildings when the estimated doses due to gaseous effluent releases from the site to areas at and beyond the site boundary would exceed 0.3 mrem to any organ over one month.

J. Explosive Gas Mixture

1. If the hydrogen concentration in the off-gas downstream of the operating recombiner reaches four percent, take appropriate action that will restore the concentration to within the limit within 48 hours.

K. Steam Jet Air Ejector (SJAE)

1. Gross radioactivity release rate from the SJAE shall be limited to less than or equal to 0.16 Ci/sec (after 30 minutes decay).

2. With the gross radioactivity release rate at the SJAE exceeding the above limit, restore the gross radioactivity release rate to within its limit within 72 hours or be in at least Hot Standby within the subsequent 12 hours.

4.8 SURVEILLANCE REQUIREMENTS

I. Ventilation Exhaust Treatment

1. See Specification 4.8.F.1 for surveillance related to AOG and Radwaste Building ventilation filter system operation.

J. Explosive Gas Mixture

1. The concentration of hydrogen in the off-gas system downstream of the recombiners shall be continuously monitored by the hydrogen monitor required operable by Table 3.9.2.

K. Steam Jet Air Ejector (SJAE)

1. The gross radioactivity release rate shall be continuously monitored in accordance with Specification 3.9.B.

2. The gross radioactivity release rate of noble gases from the SJAE shall be determined to be within the limit of Specification 3.8.K.1 at the following frequencies by performing an isotopic analysis (for Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87) on a representative sample of gases taken at the discharge.

   a. Once per week.

3.8 LIMITING CONDITIONS FOR OPERATION

3. With the gross radioactivity release rate at the SJAE greater than or equal to 1.5 Ci/sec (after 30-minute decay), restore the gross radioactivity release rate to less than 1.5 Ci/sec (after 30-minute decay), or be in Hot Standby within 12 hours.

L. Primary Containment

1. If the primary containment is to be Vented/Purged, it shall be Vented/Purged through the Standby Gas Treatment System whenever the airborne radioactivity levels in containment exceed the levels specified in 10CFR20, Appendix B, Table I, Column 1 and notes 1-5 thereto.

2. With the requirements of Specification 3.8.L.1 not satisfied, immediately suspend all Venting/Purging of the containment.

3. During normal refueling and maintenance outages when primary containment is no longer required, then Specification 3.8.G shall supersede Specifications 3.8.L.1 and 2.

4.8 SURVEILLANCE REQUIREMENTS

b. Within 4 hours following an increase of 25% or 5000 microcuries/sec, whichever is greater, in steady-state activity levels during steady-state reactor operation, as indicated by the SJAE monitor.

L. Primary Containment

1. The primary containment shall be sampled prior to venting/purging per Table 4.8.2, and if the results indicate radioactivity levels in excess of the limits of Specification 3.8.L.1, the containment shall be aligned for venting/purging through the Standby Gas Treatment System. No sampling shall be required if the venting/purging is through the Standby Gas Treatment (SBGT) System.

3.8 LIMITING CONDITIONS FOR OPERATION

M. Total Dose

1. The dose or dose commitment to a member of the public* from all station sources is limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) over a calendar year.

2. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.8.B.1.a, 3.8.B.1.b, 3.8.F.1.a, 3.8.F.1.b, 3.8.G.1.a, or 3.8.G.1.b, calculations should be made, including direct radiation contributions from the station to determine whether the above limits of Specification 3.8.M.1 have been exceeded.

4.8 SURVEILLANCE REQUIREMENTS

M. Total Dose

1. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.8.B.1, 4.8.F.1, and 4.8.G.1.

2. Cumulative dose contributions from direct radiation from plant sources shall be determined in accordance with the methods in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.8.M.2.

*NOTE: For this Specification a member of the public may be taken as a real individual accounting for his actual activities.

3.8 LIMITING CONDITIONS FOR OPERATION

N. Solid Radioactive Waste

1. The solid radwaste system shall be used in accordance with a Process Control Program as described in Section 6.12 to process wet radioactive waste (spent resins/filter sludges) to meet shipping and burial ground requirements.

2. With the provisions of Specification 3.8.N.1 not satisfied, suspend shipments of defectively processed or defectively packaged solidified wet radioactive wastes from the site.

4.8 SURVEILLANCE REQUIREMENTS

N. Solid Radioactive Waste

1. Verification of solidification of wet waste shall be performed as required and in accordance with the Process Control Program.

TABLE 4.8.1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
Batch Waste Release Tanks[b]	Prior to each release Each Batch	Prior to each release Each Batch	Principal Gamma Emitters[d]	5×10^{-7}
			I-131	1×10^{-6}
	One Batch per month sampled prior to a release	Once per month	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Prior to each release Each Batch	Once per month Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Prior to each release Each Batch	Once per quarter Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.8.1
(continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66 * S_b}{E * V * K * Y * e^{-\lambda * \Delta t}}$$

where:

LLD = the lower limit of detection as defined above (microcuries or picocuries/unit mass or volume)

S_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts/minute)

E = the counting efficiency (counts/disintegration)

V = the sample size (units of mass or volume)

K = 2.22×10^6 disintegrations/minute/microcurie or 2.22 disintegrations/minute/picocurie as applicable

Y = the fractional radiochemical yield (when applicable)

λ = the radioactive decay constant for the particular radionuclide (/minute)

Δt = the elapsed time between sample collection and analysis (minutes)

TABLE 4.8.1
(continued)

TABLE NOTATION

Typical values of E, V, Y and Δt can be used in the calculation. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples.

Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unavailable.

It should be recognized that the LLD is defined as a "before the fact" limit representing the capability of a measurement system and not as an "after the fact" limit for a particular measurement. This does not preclude the calculation of an "after the fact" LLD for a particular measurement based upon the actual parameters for the sample in question and appropriate decay correction parameters such as decay while sampling and during analysis.

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level, but as "not detected". When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.

TABLE 4.8.2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
A. Steam Jet Air Ejector	Once per week Grab Sample	Once per week	Xe-138, Xe-135, Xe-133, Kr-88, Kr-87, Kr-85M	1×10^{-4}
B. Containment Purge	Prior to each release Each Purge Grab Sample	Prior to each release Each Purge	Principal Gamma Emitters[d]	1×10^{-4}
C. Main Plant Stack	Once per month[c] Grab Sample	Once per month[c]	Principal Gamma Emitters[d]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[e]	Once per week[b] Charcoal Sample	I-131[f]	1×10^{-12}
	Continuous[e]	Once per week[b] Particulate Sample	Principal Gamma Emitters[d] (I-131)	1×10^{-11}
	Continuous[e]	Once per month Composite Particulate Sample	Gross Alpha	1×10^{-11}

TABLE 4.8.2
(continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
C. (continued)	Continuous[e]	Once per quarter Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-5}

TABLE 4.8.2
(continued)

TABLE NOTATION

a. See footnote a. of Table 4.8.1.

b. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after removal from samplers. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or thermal power change exceeding 25% of rated thermal power in one hour, and analyses shall be completed within 48 hours of changing the samples. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement to sample at least once per 24 hours for 7 days applies only if: (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has increased more than a factor of 3 and the resultant concentration is at least 1×10^{-1} uCi/ml; and (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3.

c. Sampling and analyses shall also be performed following shutdown, startup, or a thermal power change exceeding 25% of rated thermal power per hour unless: (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and the resultant concentration is at least 1×10^{-1} uCi/ml; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135 and Xe-138 for gaseous emissions, and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below LLD for the analyses should not be reported as being present at the LLD level for that nuclide, but as "not detected". When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.8.E.1, 3.8.F.1 and 3.8.G.1.

TABLE 4.8.2
(continued)

TABLE NOTATION

f. The gaseous waste sampling and analysis program does not explicitly require sampling and analysis at a specified LLD to determine the I-133 release. Estimates of I-133 releases shall be determined by counting the weekly charcoal sample for I-133 (as well as I-131) and assume a constant release rate for the release period.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents: Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site above background (at the point of discharge from the plant discharge into Connecticut River) will be less than the concentration levels specified in 10CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10CFR Part 50, to a member of the public, and (2) the limits of 10CFR Part 20.106 (e) to the population.

The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radionuclide and its MPC in air (submersion) was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

B. Liquid Effluents: Dose

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. In addition,

3.8 (Continued)

there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in potable drinking water that are in excess of the requirements of 40CFR 141. No drinking water supplies drawn from the Connecticut River below the plant have been identified. The appropriate dose equations for implementation through requirements of the Specification are described in the Vermont Yankee Off-Site Dose Calculation Manual. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I", Revision 1, April 1977.

C. Liquid Radwaste Treatment

The requirement that the appropriate portions of this system as indicated in the ODCM be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objective given in Section II.D of Appendix I to 10CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10CFR Part 50, for liquid effluents.

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest portable water supply and the nearest surface water supply in an unrestricted area.

3.8 (Continued)

E. Gaseous Effluents: Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents will be within the annual dose limits of 10CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of member(s) of the public either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 [10CFR Part 20.106(b)]. For member(s) of the public who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified limits as determined by the methodology in the ODCM, restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to (500) mrem/year to the total body or to (3,000) mrem/year to the skin.

Specification 3.8.E.b also restricts, at all times, comparable with the length of the sampling periods of Table 4.8.2 the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1500 mrem/year for the nearest cow to the plant.

F. Gaseous Effluents: Dose from Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of any member of the public through appropriate pathways is unlikely to be substantially underestimated. The appropriate dose equations are specified in the ODCM for calculating the doses due to the actual releases of radioactive noble gases in gaseous effluents. The ODCM also provides for determining the air doses at the site boundary based upon the historical average atmospheric conditions.

3.8 (Continued)

The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors,"Revision 1, July 1977.

G. Gaseous Effluents: Dose from Iodine-131, Iodine-133, Tritium, and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of the subject materials were also developed using the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I", Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors,"Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine 131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man, in areas at and beyond its site boundary. The pathways which were examined in the development of these specifications were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3.8 (Continued)

H. Gaseous Radwaste Treatment

The requirement that the appropriate portions of the Augmented Off-Gas (AOG) System be used whenever the SJAE is in operation provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objectives of Appendix I to 10CFR Part 50.

I. Ventilation Exhaust Treatment

The requirement that the AOG Building and Radwaste Building HEPA filters be used when specified provides reasonable assurance that the release of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10CFR Part 50.36a and the design objective of Appendix I to 10CFR Part 50. The requirements governing the use of the appropriate portions of the gaseous radwaste filter systems were specified by the NRC in NUREG-0473, Revision 2 (July 1979) as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10CFR Part 50, for gaseous effluents.

J. Explosive Gas Mixture

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Automatic isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled.

K. Steam Jet Air Ejector (SJAE)

Restricting the gross radioactivity release rate of gases from the main condenser SJAE provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

3.8 (Continued)

L. Primary Containment (MARK I)

This specification provides reasonable assurance that releases from containment purging/venting operations will be filtered through the Standby Gas Treatment System so that the annual dose limits of 10CFR Part 20 at the site boundary will not be exceeded. The dose objectives of Specification 3.8.G restrict purge/venting operations when the Standby Gas Treatment System is not in use and gives reasonable assurance that all releases from the plant will be kept "as low as is reasonably achievable".

M. Total Dose

This specification is provided to meet the dose limitations of 40CFR Part 190 that have been incorporated into 10CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Specific Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public is estimated to exceed the requirements of 40CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR Part 190 have not already been corrected), in accordance with the provisions of 40CFR Part 190.11 and 10CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10CFR Part 20, as addressed in Specification 3.8.A and 3.8.E. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

N. Solid Radioactive Waste

This specification implements the requirements of 10CFR Part 50.36a with respect to the handling of solid radioactive waste (spent resin and filter sludges only). The establishment and implementation of a Process Control Program (PCP), provides the operational guidelines by which proper dewatering of filter media and spent resins in preparation for off-site disposal is assured.

3.9 LIMITING CONDITIONS FOR OPERATION

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability

Applies to the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective

To assure the operability of the radioactive effluent monitoring systems and environmental programs.

Specifications

A. Liquid Effluent Instrumentation

1. During periods of release through the monitored pathway, the radioactive liquid effluent monitoring instrumentation channel shall be operable in accordance with Table 3.9.1 with their alarm setpoints set to ensure that the limits of Specification 3.8.A.1 are not exceeded.

4.9 SURVEILLANCE REQUIREMENTS

4.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

Applicability

Applies to the required surveillance of the monitoring systems or programs which perform a surveillance, protective or controlling function on the release of radioactive effluents from the plant and their identification in the environment.

Objective

To specify the type and frequency of surveillance to be applied to the radioactive effluent monitoring system and environmental programs.

Specifications

A. Liquid Effluent Instrumentation

1. Each radioactive liquid effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.1.

3.9 LIMITING CONDITIONS FOR OPERATION

B. Gaseous Effluent Instrumentation

1. The gaseous process and effluent monitoring instrumentation channels shall be operable in accordance with Table 3.9.2 with their alarm/trip setpoints set to ensure that the limits of Specifications 3.8.E.1.a, 3.8.J.1, and 3.8.K.1 are not exceeded.

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring program shall be conducted as specified in Table 3.9.3.

4.9 SURVEILLANCE REQUIREMENTS

B. Gaseous Effluent Instrumentation

1. Each gaseous process or effluent monitoring instrumentation channel shall be tested and calibrated as indicated in Table 4.9.2.

C. Radiological Environmental Monitoring Program

1. The radiological environmental monitoring samples shall be collected pursuant to Table 3.9.3 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Table 3.9.3 and the detection capabilities required by Table 4.9.3.

3.9 LIMITING CONDITIONS FOR OPERATION

D. Land Use Census

1. A land use census shall be conducted to identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles. The survey shall also identify the nearest milk animal (within 3 miles of the plant) to the point of predicted highest annual average D/Q value in each of the three major meteorological sectors due to elevated releases from the plant stack.

2. With a land use census identifying one or more locations which yield a calculated dose or dose commitment (via the same exposure pathway) at least 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.9.C.1, add the new location(s) to the radiological

4.9 SURVEILLANCE REQUIREMENTS

D. Land Use Census

1. The land use census shall be conducted at least once per year between the dates of June 1 and October 1 by either a door-to-door survey, aerial survey, or by consulting local agricultural authorities. The results of the land use census shall be included in the annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

3.9 LIMITING CONDITIONS FOR OPERATION

environmental monitoring program within 30 days if permission from the owner to collect samples can be obtained, and sufficient sample volume is available. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

E. Intercomparison Program

1. Analyses shall be performed on referenced radioactive materials supplied as part of an Intercomparison Program which has been approved by NRC.

4.9 SURVEILLANCE REQUIREMENTS

E. Intercomparison Program

1. A summary of the results of analyses performed as part of the above required Intercomparison Program shall be included in the Annual Radiological Environmental Surveillance Report. The identification of the NRC approved Intercomparison Program which is being participated in shall be stated in the ODCM.

TABLE 3.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Minimum Channels Operable	Notes
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor	1*	1,4,5
b. Service Water Discharge Monitor	1	2,4,5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Flow Rate Monitor	1*	3,4

*During releases via this pathway.

TABLE 3.9.1
(continued)

TABLE NOTATION

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.8.A.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status prior to the next release.

NOTE 5 - The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of e.8.A.1 are met during periods of release, immediately take action to suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable; or change the setpoint so it is acceptably conservative.

TABLE 3.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Notes
1. Steam Jet Air Ejector (SJAE)	1	7,8,9
a. Noble Gas Activity Monitor		
2. Augmented Off-Gas System		
a. Noble Gas Activity Monitor Between the Charcoal Bed System and the Plant Stack (Providing Alarm and Automatic Termination of Release)	1	2,5,6,7
b. Flow Rate Monitor	1	1,5,6
c. Hydrogen Monitor	1	3,5,6
3. Plant Stack		
a. Noble Gas Activity Monitor	1	2,5,7
b. Iodine Sampler Cartridge	1	4,5
c. Particulate Sampler Filter	1	4,5
d. Sampler Flow Integrator	1	1,5
e. Stack Flow Rate Monitor	1	1,5

TABLE 3.9.2
(continued)

TABLE NOTATION

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, operation of the AOG System may continue provided gas samples are collected at least once per 24 hours and analyzed within the following 4 hours, or an orderly transfer of the off-gas effluents from the operating recombiner to the standby recombiner shall be made.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment.

NOTE 5 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status within 30 days.

NOTE 6 - During releases via this pathway.

NOTE 7 - The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.8.E.1.a and 3.8.K.1 are met, immediately take actions to suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

TABLE 3.9.2
(continued)

TABLE NOTATION

Note 8 - Minimum channels operable required only during operation of the Steam Jet Air Ejector.

Note 9 - With the number of channels operable less than required by the minimum channels operable requirement, gases from the SJAE may be released to the environment for up to 72 hours provided:

1. The AOG system is not bypassed; and

2. The AOG system noble gas activity monitor is operable.

TABLE 3.9.3

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	Samples from 5 locations: 3 samples from close to the 3 site boundary locations, in different sectors, of the highest calculated annual average ground level D/Q. 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. 1 sample from a control location, as for example 15-30 km distant and in the least prevalent wind direction.	Continuous operation of sampler with sample collection semimonthly or more frequently as required by dust loading or plant effluent releases[h].	Radioiodine canister: Analyze each sample for I-131. Particulate sampler: Gross beta radioactivity analysis on each sample following filter change.[c] Composite (by location) for gamma isotopic[d] at least once per quarter.

TABLE 3.9.3
(continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION[b]	40 routine monitoring stations as follows: 16 incident response stations (one in each meteorlogical sector) within a range of 0 to 4 km8; 16 incident response stations (one in each meteorlogical sector) within a range of 2 to 8 km8; the balance of the stations to be placed in special interest areas and control station areas.	Quarterly.	Gamma dose, at least once per quarter. Incident response TLDs in the outer monitoring locations, de-dose only quarterly unless gaseous release LCO was exceeded in period.

TABLE 3.9.3
(continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface[e]	1 sample upstream.	Monthly grab sample.	Gamma isotopic analysis[d] of each sample. Tritium analysis of composite sample at least once per quarter.
	1 sample downstream.	Composite sample collected over a period of one month[f].	
b. Ground	1 sample from within 8 km distance.	Quarterly.	Gamma isotopic[d] and tritium analyses of each sample.
	1 sample from a control location.	Quarterly.	
c. Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis[d] of each sample.
	1 sample from north storm drain outfall.	As specified in the ODCM.	

TABLE 3.9.3
(continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are less than 3 primary locations available then 1 or more secondary sample from milking animals in each of 3 areas between 5 to 8 km distance where doses are calculated to be greater than 1 mrem per year.	Semimonthly if milking animals are identified on pasture; at least once per month at other times.	Gamma isotopic[d] and I-131 analysis of each sample.
	1 sample from milking animals in a control location.		
b. Fish	1 sample of two recreationally important species in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis[d] on edible portions.
	1 sample (preferably of same species) in areas not influenced by plant discharge.		
c. Vegetation	1 grass sample at each air sampling station.	Quarterly when available.	Gamma isotopic analysis[d] of each sample.
	1 silage sample at each milk sampling station (as available).	At time of harvest.	Gamma isotopic analysis[d] of each sample.

TABLE 3.9.3
(continued)

TABLE NOTATION

[a] Specific parameters of distance and direction sector from the centerline of the reactor and additional descriptions where pertinent, shall be provided for each and every sample location in Table 3.9.3 in a table and figure(s) in the ODCM. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every reasonable effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.7.C.1, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b] One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a Thermoluminescent Dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

[c] Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.9.3
(continued)

TABLE NOTATION

d Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

e The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone.

f Composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

g Each meteorological sector shall have an established "inner" and an "outer" monitoring location based on ease of recovery (i.e., response time) and year-round accessibility.

h Sample collection will be performed weekly whenever the main plant stack effluent release rate of I-131, as determined by the sampling and analysis program of Table 4.8.2, is equal to or greater than 1 x 10^{-1} uCi/sec. Sample collection will revert back to semimonthly no sooner than at least two weeks after the plant stack effluent release rate of I-131 falls and remains below 1 x 10^{-1} uCi/sec.

TABLE 3.9.4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES[a]

	Reporting Levels				
Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)
H-3	2×10^{4}[b]				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131		0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

(a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.9.4 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{concentration(1)}}{\text{reporting level(1)}} + \frac{\text{concentration(2)}}{\text{reporting level(2)}} + \ldots < 1.0.$$

When radionuclides other than those in Table 3.9.4 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Specifications 3.8.B, 3.8.E and 3.8.F.

(b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.

TABLE 4.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release				
a. Liquid Radwaste Discharge Monitor (3)	Once each day*	Prior to each release, but no more than once each month	Once each 18 months (1)	Once each quarter (2)
b. Service Water Discharge Monitor (3)	One each day	Once each month	Once each 18 months (1)	Once each quarter (2)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Discharge Flow Rate Monitor	Once each day*	Not Applicable	Not Applicable	Once each quarter*

*During releases via this pathway.

TABLE 4.9.1
(continued)

TABLE NOTATION

(1) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) liquid radioactive source positioned in a reproducible geometry with respect to the sensor. These standards shall permit calibrating the system over its normal operating range of energy and rate.

(2) The Instrument Functional Test shall also demonstrate the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicate measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(3) The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM).

TABLE 4.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Steam Jet Air Ejector (SJAE)				
a. Noble Gas Activity Monitor	Once each day**	Once each month	Once each 18 months (3)	Once each quarter (2)
2. Augmented Off-gas System				
a. Noble Gas Activity Monitor	Once each day*	Once each month	Once each 18 months (3)	Once each quarter (1)
b. Flow Rate Monitor	Once each day*	Not Applicable	Once each 18 months	Not Applicable
c. Hydrogen Monitor	Once each day*	Not Applicable	Once each quarter (4)	Once each month
3. Plant Stack				
a. Noble Gas Activity Monitor	Once each day	Once each month	Once each 18 months (3)	Once each quarter (2)
b. Sampler Flow Integrator	Once each week	Not Applicable	Once each 18 months	Not Applicable
c. System Flow Rate Monitor	Once each day	Not Applicable	Not Applicable	Not Applicable

* During releases via this pathway.
** During operation of main condenser SJAE.

TABLE 4.9.2
(continued)

TABLE NOTATION

(1) The Instrument Functional Test shall also demonstrate that automatic isolation of this pathway and the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicate measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(2) The Instrument Functional Test shall also demonstrate that Control Room alarm annunciation occurs when any of the following conditions exist:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls are not set in operate mode.

(3) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) radioactive source positioned in a reproducible geometry with respect to the sensor. These standards should permit calibrating the system over its normal operating range of rate capabilities.

(4) The Instrument Calibration shall include the use of standard gas samples (high range and low range) containing suitable concentrations, hydrogen balance nitrogen, for the detection range of interest per Specification 3.8.J.1.

TABLE 4.9.3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a][c][f]

Analysis[d]	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/Kg,wet)	Milk (pCi/l)	Vegetation (pCi/Kg,wet)	Sediment (pCi/Kg,dry)
Gross beta	4	0.01				
H-3	3000					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15[b]					
I-131		0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15[b][e]			15[b][e]		

TABLE 4.9.3
(continued)

TABLE NOTATION

(a) - See Footnote (a) of Table 4.8.1.

(b) - Parent only.

(c) - If the measured concentration minus the 5 sigma counting statistics is found to exceed the specified LLD, the sample does not have to be analyzed to meet the specified LLD.

(d) - This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the listed nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Surveillance Report pursuant to Specification 6.7.C.3.

(e) - The Ba-140 LLD and concentration can be determined by the analysis of its short-lived daughter product La-140 subsequent to an 8 day period following collection. The calculation shall be predicted on the normal ingrowth equations for a parent-daughter situation and the assumption that any unsupported La-140 in the sample would have decayed to an insignificant amount (at least 3.6 percent of its original value). The ingrowth equations will assume that the supported La-140 activity at the time of collection is zero.

(f) - Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD, but as "not detected". For purposes of averaging, the LLD will be assumed to be zero.

BASES:

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

A. Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are to ensure that the alarm will occur prior to exceeding the limits of 10CFR Part 20.

Automatic isolation function is not provided on the liquid radwaste discharge line due to the infrequent nature of batch, discrete volume, liquid discharges (on the order of once per year or less), and the administrative controls provided to ensure that conservative discharge flow rates/dilution flows are set such that the probability of exceeding the 10CFR Part 20 concentration limits are low, and the potential off-site dose consequences are also low.

B. Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

3.9 (Continued)

C. Radiological Environmental Monitoring Program

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been significantly modified as provided for by Regulatory Guides 4.3 (C.2.a) and 4.1 (C.2.b), Revision 1, April 1975. Specifically, the air particulate and radioiodine air sampling periods have been increased to semimonthly, based on plant effluent and environmental air sampling data for the previous ten years of operation. An I-131 release rate trigger value of 1×10^{-1} uCi/sec from the plant stack will require that air sample collection be increased to weekly. The 1×10^{-1} uCi/sec I-131 value corresponds to the LLD air concentration of 0.07 pCi/m^3 at the maximum predicted air monitoring station, which exhibits a maximum quarterly X/Q value of 2×10^{-7} sec/m^3. A factor of 3.5 below the LLD value has also been included in the stack release rate value to account for meteorological fluctuations in X/Q. Due to the large local population of cows and the ready availability of milk samples, food product sampling has been eliminated from the program in lieu of milk sampling. Since milking cows in the area spend very little time on pasture, silage and grass sampling have been instituted as an indicator of radionuclide deposition.

The detection capabilities required by Table 4.9.3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a before-the-fact limit representing the capability of a measurement system <u>and not</u> as an after-the-fact limit for a particular measurement. This does not preclude the calculation of an after-the-fact LLD for a particular measurement based upon the actual parameters for the sample in question.

3.9 (Continued)

D. Land Use Census

This specification is provided to ensure that changes in the use of areas at and beyond the site boundaries are identified and that modifications to the monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. The requirement of a garden census has been eliminated along with the food product monitoring requirement due to the substantial and widespread occurrence of dairy farming in the surrounding area which dominates the food uptake pathway.

The addition of new sampling locations to Specification 3.9.C, based on the land use census, is limited to those locations which yield a calculated dose or dose commitment greater than 20 percent of the calculated dose or dose commitment at any location currently being sampled. This eliminates the unnecessary changing of the environmental radiation monitoring program for new locations which, within the accuracy of the calculation, contributes essentially the same to the dose or dose commitment as the location already sampled. The substitution of a new sampling point for one already sampled when the calculated difference in dose is less than 20 percent, would not be expected to result in a significant increase in the ability to detect plant effluent related nuclides.

E. Intercomparison Program

The requirement for participation in an intercomparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

3.11 LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

1. During steady-state power operation, the MCPR Operating Limit shall be equal or greater than the values shown on Table 3.11-2. For core flows other than rated MCPR, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is given by Figure 3.11-2. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be brought to shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

Bases:

3.11C **Minimum Critical Power Ratio (MCPR)**

Operating Limit MCPR

1. The MCPR Operating Limit is a cycle-dependent parameter which can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the Fuel Cladding Integrity Safety Limit (FCISL) for potential abnormal occurrences. The MCPR operating limits are presented in Appendix A of the current cycle's Core Performance Analysis report.

5.0 DESIGN FEATURES

5.1 Site

The station is located on the property on the west bank of the Connecticut River in the Town of Vernon, Vermont, which the Vermont Yankee Nuclear Power Corporation either owns or to which it has perpetual rights and easements. The site plan showing the exclusion area boundary, boundary for gaseous effluents and boundary for liquid effluents is on Figure 2.2-5 in the FSAR. The minimum distance to the boundary of the exclusion area as defined in 10CFR100.3 is 910 feet.

No part of the site shall be sold or leased and no structure shall be located on the site except structures owned by the Vermont Yankee Nuclear Power Corporation or related utility companies and used in conjunction with normal utility operations.

5.2 Reactor

A. The core shall consist of not more than 368 fuel assemblies.

B. The reactor core shall contain 89 cruciform-shaped control rods. The control material shall be boron carbide powder (B_4C).

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.2-3 of the FSAR. The applicable design codes shall be as described in subsection 4.2 of the FSAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the FSAR.

B. The secondary containment shall be as described in subsection 5.3 of the FSAR and the applicable codes shall be as described in Section 12.0 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in subsection 5.2 of the FSAR.

d. Power Plant Design

e. Reactor Engineering

f. Radiation Safety

g. Safety Analysis

h. Instrumentation and Control

i. Metallurgy

3. Meeting Frequency: Semi-annually and as required on call of the Chairman.

4. Quorum: Chairman or Vice Chairman plus four members or designated alternates.

5. Responsibilities:

   a. Review proposed changes to the operating license including Technical Specifications.

   b. Review minutes of meetings of the Plant Operation Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

   c. Review the safety evaluations for changes to equipment or systems completed under the provisions of Section 50.59 10 CFR to verify that such actions did not constitute an unreviewed safety question.

   d. Periodic audits of implementing procedures, shall be performed under cognizance of the Committee. Included in these audits, but not limited to, are the following specific activities:

      i. plant operations;

      ii. facility fire protection program;

iii. the radiological environmental monitoring program and the results thereof at least once per 12 months;

iv. the Off-Site Dose Calculation Manual and implementing procedures at least once per 24 months;

v. the Process Control Program and implementing procedures for processing and packaging of radioactive waste at least once per 24 months;

vi. the performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

e. Investigate all reported instances of violations of Technical Specifications, reporting findings and recommendations to prevent recurrence to the Manager of Operations.

f. Perform special reviews and investigations and render reports thereon as requested by the Manager of Operations.

g. Review proposed tests and experiments and results thereof when applicable.

h. Review abnormal performance of plant equipment and anomalies.

i. Review unusual occurrences and incidents which are reportable under the provisions of 10 CFR Part 20 and 10 CFR Part 50.

j. Review of occurrences if safety limits are exceeded.

6. Authority

a. Review proposed changes to the operating license including Technical Specifications and revised bases for submittal to the NRC.

b. Review proposed changes or modifications to plant systems or equipment, provided that such changes or modifications do not involve unreviewed safety questions.

c. Recommend to the Manager of Operations appropriate action to prevent recurrence of any violations of Technical Specifications.

d. Evaluate actions taken by the Plant Operation Review Committee.

7. <u>Records</u>

Minutes of all meetings of this committee shall be recorded. Copies of the minutes shall be forwarded to the Manager of Operations, the Vice President - Operations, the Plant Manager and any others that the Chairman may designate.

6.5 Plant Operating Procedures

A. Detailed written procedures, involving both nuclear and non-nuclear safety, including applicable check-off lists and instructions, covering areas listed below shall be prepared and approved.

All procedures shall be adhered to.

1. Normal startup, operation and shutdown of systems and components of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity.

5. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.

6. Surveillance and testing requirements.

7. Fire protection program implementation including minimum fire brigade requirements and training. The training program shall meet or exceed the requirements of Section 27 of the NFPA Code 1976. Training sessions will be scheduled as plant operations permit but will be completed in specified subjects annually. Initial fire brigade training shall be completed by March 13, 1978.

8. Process Control Program in-plant implementation.

9. Off-Site Dose Calculation Manual in-plant implementation.

B. Radiation control standards and procedures shall be prepared, approved and maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR Part 20. This radiation protection program shall be organized to meet the requirements of 10 CFR Part 20.

1. Paragraph 20.203, "Caution signs, labels, signals and controls". In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

   a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

   b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them.

   c. A Health Physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who will perform periodic radiation surveillance at the frequency specified in the RWP. The surveillance frequency will be established by the Plant Health Physicist.

   The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.

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*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, providing they are following plant radiation protection procedures for entry into high radiation areas.

C. Procedures prepared for A and B above shall be reviewed and approved by the Plant Manager, or his designee, and the Manager of Operations.

D. Temporary changes to procedures described in Specification 6.5.A above which do not change the intent of the original procedure, may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented and subsequently reviewed by the PORC and approved by the Plant Manager or his designee.

E. Temporary changes to procedures described in Specification 6.5.B may be made with the concurrence of an individual holding a senior operator license and the health physicist on duty.

F. Licensed radioactive sealed sources shall be leak tested for contamination. Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement state as follows:

1. Each licensed sealed source, except startup sources previously subjected to core flux, containing radioactive materials, other than Hydrogen 3, with half-life greater than thirty days and in any form, other than gas, shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and are not being used. The sources exempted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a leak test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations.

Notwithstanding the periodic leak tests required by this Technical Specification, any licensed sealed source is exempt from such leak test when the source contains 100 microcuries or less of beta and/or gamma emitting material or 5 microcuries or less of alpha emitting material.

A special report shall be prepared and submitted to the Commission within 90 days if source leakage tests reveal the presence of $\geq$0.005 microcuries of removable contamination.

7. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

8. Records of inservice inspections of the reactor coolant system.

9. Minutes of meetings of the Plant Operation Review Committee and the Nuclear Safety Audit and Review Board.

10. Records for Environmental Qualification which are covered under the provisions of paragraph 6.9.

11. Records of analysis required by the Radiological Environmental Monitoring Program.

6.7 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Director of the appropriate Regional Office of Inspection and Enforcement unless otherwise noted.

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall, in general, include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption of commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Report

An annual report covering the previous calendar year shall be submitted prior to March 1 of each year. The annual report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurement. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the fifteenth of each month following the calendar month covered by the report. These reports shall include a narrative summary of operating experience during the report period which describes the operation of the facility and any major safety-related maintenance.

B. Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed, and reference shall be made to the original report date. Events involving systems or components described in Sections 3/4.8.B, 3/4.8.C, 3/4.8.F, 3/4.8.G, 3/4.8.H, 3/4.8.I, 3/4.8.M, 3/4.9.C, 3/4.9.D, 3/4.9.E, Table 3.9.1-note 5, Table 3.9.2-note 7, and 3/4.13 do not require reporting under the provision of this section. Such events will be reported as required in Section 6.7.C.2 or 6.7.C.3 as indicated below. The reporting provisions of this section are not applicable to Sections 3/4.8.A, 3/4.8.D, 3/4.8.E, 3/4.8.N, 3/4.9.A, and 3/4.9.B.

1/ This tabulation supplements the requirements of 20.407 of 10CFR Part 20.

1. Prompt Notification With Written Follow-Up

The types of events listed below shall be reported as expeditiously as possible, but within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate Regional Office, or his designate, no later than the first working day following the event, with a written follow-up report within two weeks. The written follow-up report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the Reactor Protection System or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the Technical specifications or failure to complete the required protective function.

Note: Instrument drift discovered as a result of testing need not be reported under this item but may be reportable under Items 1.e, 1.f or 2.a below.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition is less conservative than the least conservative aspect of the limiting condition for operation established in the Technical Specifications.

Note: If specified action is taken when a system is found to be operating between the most conservative and the least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and need not be reported under this item, but it may be reportable under Item 2.b below.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary or primary containment.

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

d. Reactivity anomalies, involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation, greater than or equal to 1% k/k; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, of subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$ or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

Note: For Items 1.e and 1.f, reduced redundancy that does not result in a loss of system function need not be reported under this section but may be reportable under Items 2.b and 2.c below.

g. Conditions arising from natural or man-made events that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence of development of an unsafe condition.

Note: This item is intended to provide for reporting of potentially generic problems.

2. Thirty Day Written Reports

The reportable occurrences discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor Protection System or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration or preventative maintenance which require system configurations as described in Items 2.a and 2.b need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in Reactor Protection Systems or Engineered Safety Feature Systems.

d. Abnormal degradation of systems other than those specified in Item 2.c above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

C. Unique Reporting Requirements

1. Semiannual Effluent Release Report

a. Within 60 days after January 1 and July 1 of each year, a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous six months of operation.

b. The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, Revision 1, June 1974, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes the format for Table 3 in Appendix B of Regulatory Guide 1.21 shall be supplemented with three additional categories: class of solid wastes (as defined by 10CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity), and solidification agent or absorbent, if any.

In addition, the radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report (or a supplement to it to be submitted within 180 days of January 1 each year) shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit during the previous calendar year. (The semiannual effluent release report submitted within 60 days of July 1 each year need not contain any dose estimates from the previous 6 months' effluent releases.) The effluent reported submitted after January 1 each year shall also include an assessment of the radiation doses from radioactive effluents to

*In lieu of submission with the first half year radioactive effluent release report, the licensee has the options of retaining this summary of required meteorological data in a file that shall be provided to the NRC upon request.

member(s) of the public due to any allowed recreational activities inside the site boundary during the previous calendar year. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. For any batch or discrete gas volume releases, the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. For radioactive materials released in continuous effluent streams, quarterly average meteorological conditions concurrent with the quarterly release period shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Off-Site Dose Calculation Manual (ODCM).

With the limits of Specification 3.8.M.1 being exceeded during the calendar year, the radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real member(s) of the public from reactor releases (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40CFR190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The radioactive effluent release reports shall include a list and description of unplanned releases from the site to site boundary of radioactive materials in gaseous and liquid effluents made during the reporting period.

With the quantity of radioactive material in any outside tank exceeding the limit of Specification 3.8.D.1, describe the events leading to this condition in the next Radioactive Effluent Release Report.

If inoperable radioactive liquid effluent monitoring instrumentation is not returned to operable status prior to the next release pursuant to Note 4 of Table 3.9.1, explain in the next Radioactive Effluent Release Report the reason(s) for delay in correcting the inoperability.

If inoperable gaseous effluent monitoring instrumentation is not returned to operable status within 30 days pursuant to Note 5 of Table 3.9.2, explain in the next Radioactive Effluent Release Report the reason(s) for delay in correcting the inoperability.

With milk samples no longer available from one or more of the sample locations required by Table 3.9.3, identify the cause(s) of the sample(s) no longer being available, identify the new location(s) for obtaining available replacement samples, and include revised ODCM figure(s) and table(s) reflecting the new location(s) in the next Radioactive Effluent Release Report.

With a land use census identifying one or more locations which yield at least a 20 percent greater dose or dose commitment than the values currently being calculated in Specification 4.8.G.1, identify the new location(s) in the next Radioactive Effluent Release Report.

Changes made during the reporting period to the Process Control Program (PCP) and to the Off-Site Dose Calculation Manual (ODCM), shall be identified in the next Radioactive Effluent Release Report.

2. <u>Special Reports</u>

Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

a. Liquid Effluents, Specifications 3.8.B and 3.8.C.

With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the limits of Specification 3.8.B.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.B.1.

With liquid radwaste being discharged without processing through appropriate treatment systems and estimated doses in excess of Specification 3.8.C.1, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

(3) Summary description of action(s) taken to prevent a recurrence.

b. Gaseous Effluents, Specifications 3.8.F, 3.8.G, 3.8.H, and 3.8.I.

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the limits of Specification 3.8.F.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.F.1. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and/or radionuclides in particulate form exceeding any of the limits of Specification 3.8.G.1, prepare and submit to the Commission within 30 days a special report which identifies the cause(s) for exceeding the limit(s) and the corrective action(s) taken to assure that subsequent releases will be in compliance with the limits of Specification 3.8.G.1.

With gaseous radwaste being discharged without processing through appropriate treatment systems as defined in Specification 3.8.H.1 for more than seven (7) consecutive days, or in excess of the limits of Specification 3.8.I.1, prepare and submit to the Commission within 30 days a special report which includes the following information:

(1) explanation of why gaseous radwaste was being discharged without treatment (Specification 3.8.H.1), or with resultant doses in excess of Specification 3.8.I.1, identification of any inoperable equipment or subsystems, and the reasons for the inoperability;

(2) action(s) taken to restore the inoperable equipment to operable status; and

(3) summary description of action(s) taken to prevent a recurrence.

c. Total Dose, Specification 3.8.M.

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Specification 3.8.M, prepare and submit to the Commission within 30 days a special report which defines the corrective action(s) to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.8.M and includes the schedule for achieving conformance with these limits. This special report, required by 10CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a member of the public from station sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated doses exceed any of the limits of Specification 3.8.M, and if the release condition resulting in violation of 40CFR Part 190 has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

d. Radiological Environmental Monitoring, Specification 3.9.C.

With the level of radioactivity as the result of plant effluents in an environmental sampling media at one or more of the locations specified in Table 3.9.3 exceeding the reporting levels of Table 3.9.4, prepare and submit to the Commission within 30 days from the receipt of the Laboratory Analyses a special report which includes an evaluation of any release conditions, environmental factors or other factors which caused the limits of Table 3.9.4 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents, however, in such an event, the condition shall be reported and described in the annual Radiological Environmental Surveillance Report.

e. Land Use Census, Specification 3.9.D.

With a land use census not being conducted as required by Specification 3.9.D, prepare and submit to the Commission within 30 days a special report which identifies the reasons why the survey was not conducted, and what steps are being taken to correct the situation.

3. Environmental Radiological Monitoring

Radiological Environmental Surveillance Reports covering the operation of the unit during previous calendar year shall be submitted prior to May 1 of each year.

The annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment.

The annual Radiological Environmental Surveillance Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to the table and figures in the ODCM. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

With the level of radioactivity in an environmental sampling media at one or more of the locations specified in Table 3.9.3 exceeding the reporting levels of Table 3.9.4, the condition shall be described in the next annual Radiological Environmental Surveillance Report only if the measured level of radioactivity was not the result of plant effluents. With the radiological environmental monitoring program not being conducted as specified in Table 3.9.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence shall be included in the next annual Radiological Environmental Surveillance Report.

The annual Radiological Environmental Surveillance Report shall also include the results of the land use census required by Specification 3.9.D. A summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances and directions from the reactor shall be in the reports. If new environmental sampling locations are identified in accordance with Specification 3.9.D, the new locations shall be identified in the next annual Radiological Environmental Surveillance Report.

The reports shall also include a discussion of all analyses in which the LLD required by Table 4.9.3 was not achievable.

The results of licensee participation in the intercomparison program required by Specification 3.9.E shall be included in the reports. With analyses not being performed as required by Specification 3.9.E, the corrective actions taken to prevent a recurrence shall be report to the Commission in the next annual Radiological Environmental Surveillance Report.

6.8 Fire Protection Inspection

A. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm.

B. An inspection and audit by an outside fire consultant shall be performed at intervals no greater than 3 years.

6.9 Environmental Qualification

A. By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of Division of Operating Reactors, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-28, dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise qualified.

6.10 Integrity of Systems Outside Containment

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels will be implemented. This program shall include the following:

A. Provisions establishing preventive maintenance and periodic visual inspection requirements.

B. System leakage inspections, to the extent permitted by system design and radiological conditions, for each system at a frequency not to exceed refueling cycle intervals. The systems subject to this testing are: (1) Residual Heat Removal, (2) Core Spray, (3) Reactor Water Cleanup, (4) HPCI, (5) RCIC, and (6) Sampling Systems.

6.11 Iodine Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas* under accident conditions will be implemented. This program shall include the following:

A. Training of personnel.

B. Procedures for monitoring.

C. Provisions for maintenance of sampling and analysis equipment.

6.12 Process Control Program (PCP)

A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

A. Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semiannual Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

a. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information.

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*Areas requiring personnel access for establishing hot shutdown condition.

b. A determination that the change did not reduce the overall conformance of the dewatered spent resins/filter media waste product to existing criteria for solid waste shipments and disposal.

c. Documentation of the fact that the change has been reviewed by PORC and approved by the Manager of Operations (MOO).

2. Shall become effective upon review by PORC and approval by the Manager of Operations (MOO).

6.13 Off-Site Dose Calculation Manual (ODCM)

An Off-Site Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

A. Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the semiannual Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s).

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

c. Documentation of the fact that the change has been reviewed by PORC and approved by the Manager of Operations (MOO).

2. Shall become effective upon review by PORC and approved by the Manager of Operations (MOO).

6.14 Major Changes to Radioactive Liquid, Gaseous, and Solid Waste Treatment Systems*

Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

A. Shall be reported to the Commission in the semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10CFR Part 50.59;

2. Sufficient detailed information to support the reason for the change without benefit of additional or supplemental information;

3. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to member(s) of the public at the site boundary and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by PORC.

B. Shall become effective upon review and acceptance by PORC and approval by the Plant Manager.

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* Licensee may choose to submit the information called for in this Specification as part of the annual FSAR update.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 83 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, Vermont Yankee Nuclear Power Corporation (VYNPC) has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. VYNPC filed this information with the Commission by letter dated January 23, 1984, which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-28 for Vermont Yankee Nuclear Power Station. The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0473. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors", provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of Part 20.106, "Radioactivity in Effluents to Unrestricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases for radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

10 CFR Part 50, Appendix I, Section IV, provides guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0473 provides radiological effluent technical specifications for boiling water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment sytems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent technical specifications may be acceptable if the staff determines that the alternatives are in compliance with the regulations and with the intent of the regulatory guidance.

The standard radiological effluent technical specifications can be grouped under the following categories:

(1) Instrumentation

(2) Radioactive effluents

(3) Radiological environmental monitoring

(4) Design features

(5) Administrative controls.

Each of the specifications under the first three categories is comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions of operation or surveillance requirements.

Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.D.

3.0 EVALUATION

The enclosed report (TER-C5506-116) was prepared for us by Franklin Research Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the Licensee's submittal with NRC provided criteria. The staff has reviewed this TER and agrees with the evaluation. A copy of the TER is enclosed.

4.0 SUMMARY

The proposed changes to the radiological effluent technical specifications for Vermont Yankee Nuclear Power Station have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0473 (Vermont Yankee is a boiling water reactor) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

1. Relation B[illegible] Provisions of the Regulations and the Standard Radiol[illegible] Effluent Technical Specifications for Pressurized Water[illegible] tors and Boil[illegible] ater Reactors

Indicate the specifications that are needed to assure compliance with the identified provision of the regulations.

Standard Radiological Effluent Technical Specifications	Instrumentation		Radioactive Effluents: Liquid				Gaseous: PWR/BWR				Gaseous: PWR		Gaseous: BWR						Rad. Envir. Monitoring			Design Features	Administrative Control						
visions of Title 10 Code of Federal Regulations	Rad. Liquid Effl. Monitoring	Rad. Gas. Effl. Monitoring	Effluent Concentration	Dose	Liquid Radwaste Treatment	Liquid Holdup Tanks	Dose Rate	Dose Noble Gases	Dose I-131, Trit., and Part.	Explosive Gas Mixture	Gaseous Radwaste Treatment	Gas Storage Tanks	Gaseous Radwaste Treatment	Ventilation Exhaust Treatment	Main Condenser	Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Site Boundaries*	Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
.36a Technical specifications on effluents from nuclear power reactors																													
nain within limits of § 20.106			●			●	●					●				●												●	
tablish and follow procedures to control ffluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				●	
intain and use radioactive waste system quipment				●	●			●	●		●		●	●														●	
bmit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●	●
0.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●							●	
50 Appendix A - General Design Criteria																													
iterion 60 - Control of releases of radioactive materials to the environment	●	●			●	●				●	●	●	●	●	●	●	●							●			●	●	
iterion 61 - Fuel storage and handling and radioactivity control	●	●																											
iterion 63 - Monitoring fuel and waste storage	●	●																											
iterion 64 - Monitoring radioactivity releases	●	●													●				●									●	
50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●		
50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																													
intain releases within design objectives				●	●			●	●		●			●														●	
tablish surveillance & monitoring program to rovide data on:																													
(1) quantities of rad. matls. in effluents	●	●																											
(2) radiation & rad. matls. in the environment																			●						●	●			
(3) changes in use of unrestricted areas																				●									
ert best efforts to keep releases "ALARA"				●	●			●	●		●		●	●															●
bmit report if calculated doses exceed the lesign objective				●				●	●		●			●											●				
monstrate conform. to des. obj. by calc. proced.				●				●	●		●			●														●	
100															●														

e: Needed to fully implement other specifications.

5.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 GENERAL CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Meinke

Attachment:
EG&G Technical Evaluation
Report

Dated: October 9, 1984

TECHNICAL EVALUATION REPORT

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION IMPLEMENTATION (A-2)

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

NRC DOCKET NO. 50-271

NRC TAC NO. 8129

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 4

FRC TASK 116

Prepared by

Franklin Research Center
20th and Race Streets
Philadelphia, PA 19103

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Author: S. Chen

FRC Group Leader: S. Pandey

Lead NRC Engineer: F. Congel
C. Willis

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Prepared by:	Reviewed by:	Approved by:
S. Chen	S. Pandey	T. Carfagno
Principal Author	Group Leader	Department Director
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Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of Vermont Yankee Nuclear Power Station with regard to Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

The evaluation uses criteria proposed by the NRC staff in the Model Technical Specifications for boiling water reactors (BWRs), NUREG-0473 [1]. This effort is directed toward the NRC objective of implementing RETS which comply principally with the regulatory requirements of the Code of Federal Regulations, Title 10, Part 50 (10CFR50), "Domestic Licensing of Production and Utilization Facilities," Appendix I [2]. Other regulations pertinent to the control of effluent releases are also included within the scope of compliance.

1.2 GENERIC BACKGROUND

Since 1970, 10CFR50, Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," has required licensees to provide technical specifications which ensure that radioactive releases will be kept as low as reasonably achievable (ALARA). In 1975, numerical guidance for the ALARA requirement was issued in 10CFR50, Appendix I [3]. The licensees of all operating reactors were required to submit, no later than June 4, 1976, their proposed ALARA Technical Specifications and information for evaluation in accordance with 10CFR50, Appendix I.

However, in February 1976, the NRC staff recommended that proposals to modify Technical Specifications be deferred until the NRC completed the model RETS. The model RETS deals with radioactive waste management systems and environmental monitoring. Although the model RETS closely parallels 10CFR50, Appendix I requirements, it also includes provisions for addressing other issues.

These other issues are specifically stipulated by the following regulations:

- o 10CFR20 [4], "Standards for Protection Against Radiation," Paragraphs 20.105(c), 20.106(g), and 20.405(c) require that nuclear power plants and other licensees comply with 40CFR190 [5], "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40CFR190 limits have been or may be exceeded.
- o 10CFR50, Appendix A [6], "General Design Criteria for Nuclear Power Plants," contains Criterion 60 - Control of releases of radioactive materials to the environment; Criterion 63 - Monitoring fuel and waste storage; and Criterion 64 - Monitoring radioactivity releases.
- o 10CFR50, Appendix B [7], establishes the quality assurance required for nuclear power plants.

The NRC position on the model RETS was established in May 1978 when the NRC's Regulatory Requirements Review Committee approved the model RETS: NUREG-0473 [1] for BWRs and NUREG-0472 for pressurized water reactors (PWRs) [8]. Copies were sent to licensees in July 1978 with a request to submit proposed site-specific RETS on a staggered schedule over a 6-month period. Licensees responded with requests for clarifications and extensions.

The Atomic Industrial Forum (AIF) formed a task force to comment on the model RETS. NRC staff members first met with the AIF task force on June 17, 1978. The model RETS was subsequently revised to reflect comments from the AIF and others. A principal change was the transfer of much of the material concerning dose calculations from the model RETS to a separate ODCM.

The revised model RETS was sent to licensees on November 15 and 16, 1978 with guidance (NUREG-0133 [9]) for preparation of the RETS and the ODCM and a new schedule for responses, again staggered over a 6-month period.

Four regional seminars on the RETS were conducted by the NRC staff during November and December 1978. Subsequently, Revision 2 of the model RETS and additional guidance on the ODCM were issued in February 1979 to each utility at individual meetings. In response to the NRC's request, operating reactor licensees subsequently submitted initial proposals on plant RETS and the ODCM. Review leading to ultimate implementation of these documents was initiated by the NRC in 1981 using subcontracted independent teams as reviewers.

As the RETS review process has progressed since September 1981, feedback from the licensees has led the NRC to believe that modification to some of the guidelines in the current version of Revision 2 is needed to clarify specific concerns of the licensees and thus expedite the entire review process. Starting in April 1982, NRC distributed revised versions of RETS in draft form to the licensees during site visits. The new guidance on these changes was presented at the AIF meeting on May 19, 1982 [10]. Some interim changes regarding the Radiological Environmental Monitoring Section were issued in 1982 [11, 12]. With the incorporation of these new changes, NRC issued, in December 1983, a draft version of NUREG-0473, Revision 3 [13], to serve as new guidance for the review teams.

1.3 PLANT-SPECIFIC BACKGROUND

In response to the NRC's request, the Licensee, Vermont Yankee Nuclear Power Corporation (VYNPC), submitted a RETS proposal dated February 3, 1979 [14] on behalf of Vermont Yankee Nuclear Power Station, which was followed by a submittal of the ODCM [15]. In the RETS submittal, the Licensee had used non-standard format. In an initial evaluation by the Franklin Reseach Center (FRC), an independent review team, the Licensee's RETS submittal was evaluated against the model RETS (NUREG-0473, Draft Revision 3) and assessed for compliance with the stipulated provisions. Review of the ODCM was conducted in accordance with NRC-issued guidelines (NUREG-0133). Copies of the draft review, dated August 20, 1982 [16, 17], were delivered to the NRC and the Licensee prior to a site visit by the reviewers.

The site visit was conducted on September 16-17, 1982 by the reviewers with the participation of plant personnel and the NRC staff. Discussions focused on the initial review of the proposed changes to the RETS and on the technical approaches for an ODCM. The deficiencies in the Licensee's proposed RETS were considered, deviations from NRC guidelines were pointed out, many differences were clarified, and only a few items remained unresolved pending justification by the Licensee. These issues are summarized in Reference 18.

In a letter transmittal dated January 24, 1983 [19], the Licensee sent FRC a revised draft RETS for review. In this submittal, the RETS was written in a non-standard format. Also included in the package was a list of justifications for major deviations from the model RETS. The Licensee also submitted a draft ODCM to FRC for review on July 14, 1983 [20]. Both draft submittals were reviewed by FRC, and discrepancies were documented [21, 22] and transmitted to NRC.

The final version of the Vermont Yankee RETS [23], dated December 1983, was submitted to the NRC and transmitted to the FRC reviewers together with justifications provided by the Licensee. The document was subsequently reviewed. Final evaluation of RETS was detailed in a comparison report [24] which used NUREG-0473, Draft Revision 3 [12] to evaluate the Licensee's submittal. The comparison report also incorporates NRC comments [25, 26] which serve as additional guidelines regarding plant-specific issues.

2. REVIEW CRITERIA

Review criteria for the RETS and ODCM were provided by the NRC in three documents:

NUREG-0472, RETS for PWRs

NUREG-0473, RETS for BWRs

NUREG-0133, Preparation of RETS for Nuclear Power Plants.

Twelve essential criteria are given for the RETS and ODCM:

1. All significant releases of radioactivity shall be controlled and monitored.

2. Offsite concentrations of radioactivity shall not exceed the 10CFR20, Appendix B, Table II limits.

3. Offsite radiation doses of radioactivity shall be ALARA.

4. Equipment shall be maintained and used to keep offsite doses ALARA.

5. Radwaste tank inventories shall be limited so that failures will not cause offsite doses exceeding 10CFR20 limits.

6. Hydrogen and/or oxygen concentration in the waste gas system shall be controlled to prevent explosive mixtures.

7. Wastes shall be processed to shipping and burial ground criteria under a documented program, subject to quality assurance verification.

8. An environmental monitoring program, including a land-use census and an interlaboratory comparison program, shall be implemented.

9. The radwaste management program shall be subject to regular audits and reviews.

10. Procedures for control of liquid and gaseous effluents shall be maintained and followed.

11. Periodic and special reports on environmental monitoring and on releases shall be submitted.

12. Offsite dose calculations shall be performed using documented and approved methods consistent with NRC methodology.

Subsequent to the publication of NUREG-0472 and NUREG-0473, the NRC staff issued guidelines [27, 28], clarifications [29, 30], and branch positions [31, 32, 33, 34] establishing a policy that guides the licensees of operating reactors to meet the intent, if not the letter, of the model RETS provisions. The NRC branch positions issued since the RETS implementation review began have clarified the model RETS implementation for operating reactors.

Review of the ODCM was based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual" [35]; NUREG-0133 [9]; and Regulatory Guide 1.109 [36]. The ODCM format is left to the licensee and may be simplified by tables and grid printouts.

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF RADIOLOGICAL EFFLUENT SYSTEM

This section briefly describes the liquid and gaseous effluent radwaste treatment systems, release paths, and control systems installed at Vermont Yankee Nuclear Power Station, a BWR.

3.1.1 Radioactive Liquid Effluent

The liquid radioactive wastes at the Vermont Yankee plant consist of four categories: high purity wastes, low purity wastes, chemical wastes, and detergent wastes. The high purity wastes are processed by filtration and ion exchange through the waste collector filter or fuel pool and waste demineralizer as required. Following processing, the liquid is pumped to the waste sample tank where it is sampled. If high purity requirements are met, the waste contents are normally transferred to the condensate storage tank. Otherwise, the liquid wastes are either recycled through the radwaste system or discharged. The low purity wastes, chemical wastes, and detergent wastes generally have a low concentration of radioactive impurities, and processing consists of simple filtration. If necessary, these wastes can be combined with the high purity wastes for subsequent processing. All wastes are discharged in batches into the circulating water discharge structure (see Figure 1) for release to the Connecticut River. Also joined into the circulating water is the continuous discharge of service water.

3.1.2 Radioactive Gaseous Effluent

The process gases from the Vermont Yankee Nuclear Power Station are routed to the plant stack for dilution and elevated release for discharge to the atmosphere (see Figure 2). The substreams routed to the plant stack are the building ventilation, the turbine gland seal/mechanical vacuum pumps, and the standby gas treatment system. The gaseous radwaste system includes the augmented offgas subsystem (AOG) for the main condenser air ejectors and other

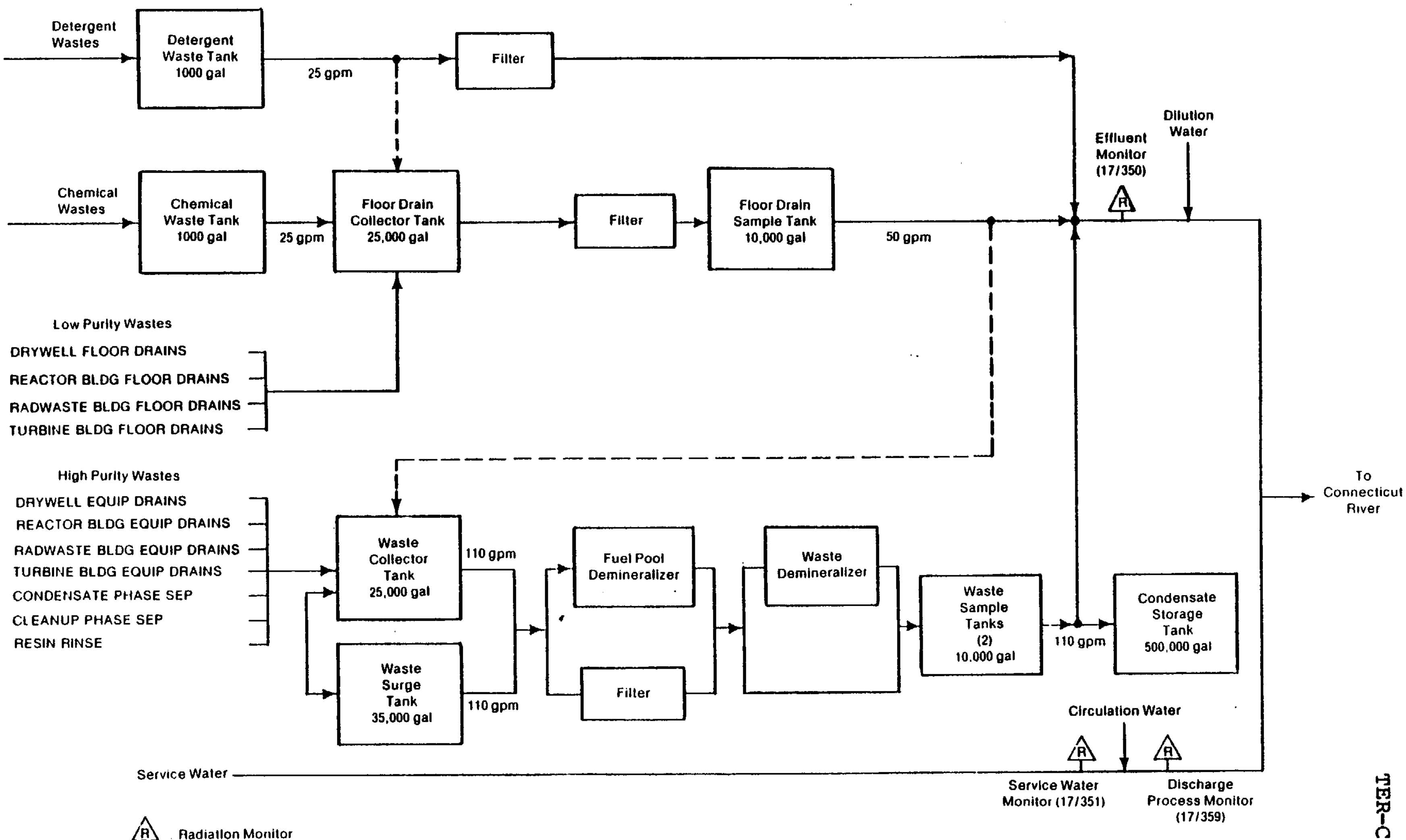

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for Vermont Yankee Nuclear Power Station

TER-C5506-116

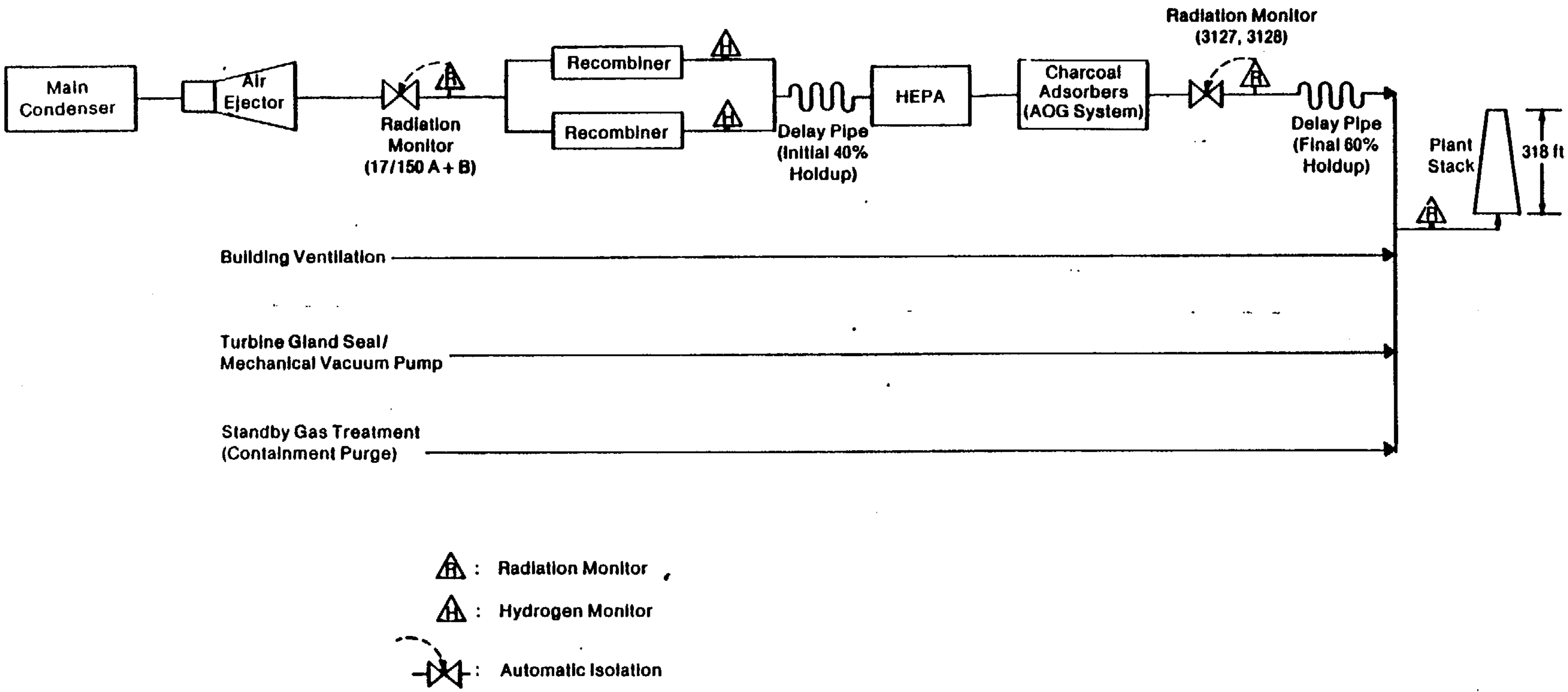

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for Vermont Yankee Nuclear Power Station

subsystems for the startup vacuum pump and the gland seal condenser. The AOG system consists of a dual hydrogen dilution and recombiner subsystem and a single charcoal adsorber subsystem.

3.2 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

The evaluation of the Licensee's proposed RETS against the provisions of NUREG-0473 included the following: (1) a review of information provided in the Licensee's 1979 and 1983 draft submittals [14, 15, 19, 20], (2) the resolution of problem areas in those submittals by means of a site visit [18], and (3) a review of the Licensee's December 1983 RETS submittal [23].

3.2.1 Effluent Instrumentation

The objective of the RETS with regard to effluent instrumentation is to ensure that all significant releases of radioactivity are monitored. The RETS specify that all effluent monitors be operable and alarm/trip setpoints be determined to ensure that radioactivity levels do not exceed the maximum permissible concentration (MPC) set by 10CFR20. To further ensure that the instrumentation functions properly, surveillance requirements are needed in the specifications.

3.2.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

A radiation monitor (No. 17/350) has been installed for the liquid radwaste effluent line (Figure 1) which combines effluent streams from the releases of high purity wastes, low purity wastes, chemical wastes, and detergent wastes. The Licensee has also provided a monitor (No. 17/351) for the main service water, which combines with the circulation water at the discharge structure, where a process monitor (No. 17/359) is also provided.

The Licensee has not provided automatic isolation function for the liquid radwaste effluent line. However, because of the infrequent nature of the batch discharges (on the order of once per year or less) and the adequate sampling program provided by the Licensee, the instrumentation monitoring system is deemed to meet the intent of NUREG-0473.

These existing monitoring capabilities have provided adequate assurance that the provisions of NUREG-0473 for the radioactive liquid effluent monitoring instrumentation are met.

3.2.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation

The plant main stack is provided with a monitoring system capable of monitoring noble gases, iodines, and particulates. The main condenser air ejector for each unit also has a redundant noble gas monitor (No. 17/150 A and B). The Licensee has also provided redundant radiation monitors (No. 3127, 3128) for the AOG treatment system. This AOG monitor is equipped with automatic isolation capability as specified by NUREG-0473.

The existing monitoring capabilities provided by the Licensee have met the intent of NUREG-0473 for radioactive gaseous effluent monitor instrumentation.

3.2.2 Concentration and Dose Rates of Effluents

3.2.2.1 Liquid Effluent Concentration

In Section 3.8.A of the Licensee's submittal, a commitment is made to maintain the concentration of radioactive liquid effluents released to unrestricted areas to within 10CFR20 limits, and, if the concentration of liquid effluents exceeds these limits, the concentration will be restored immediately to a value equal to or less than the MPC specified in 10CFR20. All batches of radioactive liquid effluents from the release tanks are sampled and analyzed in accordance with a sampling and analysis program (Table 4.8.1 of the Licensee's submittal) which meets the intent of NUREG-0473.

It was determined that the Licensee-proposed alternative meets the intent of NUREG-0473.

3.2.2.2 Gaseous Effluent Dose Rate

In Section 3.8.E of the Licensee's submittal, a commitment is made to maintain the offsite dose rate from radioactive gaseous effluents to areas at and beyond the site boundary within 10CFR20 limits, or the equivalent dose rate values prescribed by Section 3.11.2.1 of NUREG-0473. If the dose rate of gaseous effluents exceeds these limits, it will be restored immediately to a value equal to or less than these limits. This commitment satisfies the provisions of NUREG-0473.

The radioactive gaseous waste sampling and analysis program (Table 4.8.2 of the Licensee's submittal) provides adequate sampling and analysis of the plant stack discharges, including the substreams, and therefore meets the intent of NUREG-0473.

3.2.3 Offsite Doses from Effluents

The objective of the RETS with regard to offsite doses from effluents is to ensure that offsite doses are kept ALARA and are in accordance with 10CFR50, Appendix I, and 40CFR190. The Licensee has made a commitment to (1) meet the quarterly and yearly dose limitations for liquid effluents, per Section II.A of Appendix I, 10CFR50; (2) restrict the air doses for beta and gamma radiation from the site to areas at and beyond the site boundary as specified in 10CFR50, Appendix I, Section II.B; (3) maintain the dose level at and beyond the site boundary from release of iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days within the design objectives of 10CFR50, Appendix I, Section II.C; and (4) limit the annual dose from all station sources of the plant to any member of the public to within the requirements of 40CFR190. In each pertinent section, the Licensee has made a commitment to perform dose calculations in accordance with methods given in the ODCM. This satisfies the intent of NUREG-0473.

3.2.4 Effluent Treatment

The objectives of the RETS with regard to effluent treatment are to ensure that wastes are treated to keep releases ALARA and to satisfy the provisions of Technical Specifications governing the maintenance and use of radwaste treatment equipment. The Licensee has made a commitment to use the liquid (Section 3.8.C of the Licensee's submittal) radwaste treatment systems when the projected dose, averaged over 31 days, exceed 25% of the annual dose design objectives, prorated monthly. For gaseous radwaste, the Licensee proposes to treat the effluents by operating the gaseous radwaste treatment system (AOG system) whenever the main condenser air ejector system is in operation. The Licensee's existing gaseous radwaste treatment system also includes the offgas holdup line, HEPA filters, and the stack filter house filtration. The Licensee has also made a commitment to use the ventilation exhaust treatment system when the projected monthly dose exceeds the limit specified by NUREG-0473. The Licensee has also made a commitment in the ODCM to calculate the projected doses on a monthly basis. It is determined that the Licensee's proposal meets the intent of NUREG-0473.

3.2.5 Radioactivity Inventory Limits

The objective of the RETS with regard to the liquid tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10CFR20 for non-occupational exposure. Also, the gaseous radioactivity release inventory is to be limited to within a rate of 100 microcuries per second per megawatt thermal, during the operation of the main condenser air ejector. The Licensee has provided a limit of 10 curies for any outside temporary tanks. For radioactivity releases from the main condenser air ejector, a release rate limit of 0.16 curies/sec (after 30 min delay) has been set for noble gases, which is based on the rated thermal power of 1593 MWt at Vermont Yankee Nuclear Power Station. The Licensee has also made a commitment to take a more immediate action, i.e., to place the plant in hot standby within 12 hours, if the release rate from the air ejector exceeds 1.5 curies/sec. The Licensee's commitment to comply with these radioactivity inventory and release rate limits satisfies the intent of NUREG-0473.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in waste gas systems. The Licensee claimed that the waste gas system is designed to withstand a hydrogen explosion and has thus made a commitment to maintain a safe concentration of hydrogen in the main condenser offgas treatment system using redundant hydrogen monitoring systems for recombiner trains. This commitment satisfies the intent of NUREG-0473.

3.2.7 Solid Radwaste System

The objective of the RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before it is shipped to the burial site. Specification 3.11.3 of NUREG-0473 provides for the establishment of a PCP, or the equivalent, to show compliance with this objective. The Licensee has made a commitment to implement such a program in accordance with a PCP and to thus ensure that radwaste is properly processed and packaged before it is shipped to the burial site. This meets the intent of NUREG-0473.

3.2.8 Radiological Environmental Monitoring Program

The objectives of the RETS with regard to environmental monitoring are to ensure that an adequate and full-area-coverage environmental monitoring program exists and that the 10CFR50, Appendix I requirements for technical specifications on environmental monitoring are satisfied. In all cases, the Licensee has followed NUREG-0473 guidelines, including the Branch Technical Position dated November 1979 [32], and has provided an adequate number (40) of sample locations for pathways identified. The Licensee's methods of analysis and maintenance of yearly records satisfy the NRC guidelines and meet the intent of 10CFR50, Appendix I. The Licensee has also made a commitment to document the environmental monitoring sample locations in the ODCM, which meets the intent of NUREG-0473. The specification for the land use census satisfies the provisions of Section 3.12.2 of NUREG-0473 by providing for an annual census in the specified areas. The Licensee participates in an interlaboratory comparison program approved by the NRC and reports the results in the Annual Radiological Environmental Operating Report, which also meets the intent of NUREG-0473.

3.2.9 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The Licensee's administrative structure designates the Plant Operations Review Committee (PORC) and the Nuclear Safety Audit and Review Committee (NSARC) as the groups responsible for the review and audit of the radiological environmental monitoring program, the ODCM, and the PCP. The PORC is responsible for reviewing the procedures associated with these programs. The NSARC is responsible for auditing the program as often as is specified under NUREG-0473.

3.2.10 Procedures and Records

The objective of the RETS with regard to procedures is to satisfy the provisions for written procedures specified in NUREG-0473 for implementing the ODCM, the PCP, and the quality program (QA) program. It is also an objective of RETS to properly retain the documented records related to the environmental monitoring program and certain QA procedures. The Licensee has made a commitment to establish, implement, and maintain written procedures for the PCP, ODCM, and QA program according to the provisions of NUREG-0473 [13]. The Licensee intends to retain the records of the radiological environmental monitoring program for the duration of the facility operating license. It is determined that the Licensee has met the intent of NUREG-0473 in these areas.

3.2.11 Reports

In addition to the reporting requirements of Title 10, Code of Federal Regulations (10CFR), the objective of the RETS with regard to administrative controls is also to ensure that appropriate periodic and special reports are submitted to the NRC.

The Licensee made a commitment to follow applicable reporting requirements stipulated by 10CFR regulations and also the following reports specified by NUREG-0473:

1. Annual radiological environmental operating report. In Section 6.7.C.3 of the Licensee's submittal, a commitment was made to provide an annual radiological environmental surveillance report that includes summaries, interpretations, and analysis of the results of the environmental surveillance activities. The report also includes the results of land use censuses, and participation in an interlaboratory comparison program specified by Specification 3.12.3 of NUREG-0473.

2. Semiannual radioactive release reports. In Section 6.7.C.1 of the Licensee's submittal, a commitment was made to provide semiannual effluent release reports which include a summary of radioactive liquid and gaseous effluents and solid waste released, an assessment of offsite doses, and a list of unplanned releases. Listing of new locations for dose calculations identified by the land use census as well as any changes to ODCM and PCP are also included in the report.

3. Special report. In Section 6.7.C.2 of the Licensee's submittal, a commitment was made to file a 30-day special report to the NRC under the following conditions as prescribed by the proposed specifications:

   o Exceeding radioactive liquid effluent limits according to:

     Dose, Specification 3.8.B.1
     Liquid Waste Treatment, Specification 3.8.C.1

   o Exceeding radioactive gaseous effluent limits according to:

     Dose, Specifications 3.8.F.1 and 3.8.G.1
     Gaseous Waste Treatment, Specifications 3.8.H.1 and 3.8.I.1

   o Exceeding radioactive effluent limits according to:

     Uranium Fuel Cycle Dose Commitment, Specification 3.8.M

   o Exceeding the reporting levels of Table 3.9.4 for the radioactivity measured in the environmental sampling medium, Specification 3.9.C

   o Land use census not being conducted in accordance with Specification 3.9.D.

These reporting commitments have satisfied the provisions of NUREG-0473.

3.2.12 Implementation of Major Programs

One objective of the administrative controls is to ensure that implementation of major programs such as the ODCM, PCP, and major changes to the radioactive waste treatment system follow appropriate administrative procedures. The Licensee has made a commitment to review, report, and implement major programs such as the ODCM, PCP, and major changes to the radioactive waste treatment system. This commitment meets the intent of NUREG-0473.

3.3 OFFSITE DOSE CALCULATION MANUAL (ODCM)

As specified in NUREG-0473, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and maintain the operability of the effluent systems. As a minimum, the ODCM should provide equations and methodology for the following topics:

- alarm and trip setpoint on effluent instrumentation
- liquid effluent concentration in unrestricted areas
- gaseous effluent dose rate at or beyond the site boundary
- liquid and gaseous effluent dose contributions
- liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams, consistent with the systems being used at the station, defining the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. Of course, these diagrams should be consistent with the systems being used at the station. A description and location of samples in support of the environmental monitoring program are also needed in the ODCM.

3.3.1 Evaluation

The Licensee has followed the methodology of NUREG-0133 [9] to determine the alarm and trip setpoints for the liquid and gaseous effluent monitors, which ensures that the maximum permissible concentrations, as specified in 10CFR20, will not be exceeded by discharges from various liquid or gaseous release points.

The Licensee demonstrated the method of calculating the radioactive liquid concentration by describing in the ODCM the means of collecting and analyzing representative samples prior to and after releasing liquid effluents into the circulating water discharge. The method provides added assurance of compliance with 10CFR20 for liquid effluent releases.

Methods are also included for showing that dose rates at or beyond the site boundary due to noble gases, iodine-131, tritium, and particulates with half-lives greater than 8 days are in compliance with 10CFR20. In this calculation, the Licensee has considered effluent releases from the plant stack; those releases are being treated as elevated level. In all cases, the Licensee has used the highest annual average values of relative concentration (X/Q) and relative deposition (D/Q) to determine the controlling locations. For elevated releases from the main stack, the Licensee has also considered the direct radiation contribution from exposure to the finite plume. The Licensee intends to use the maximally exposed individual and the critical organ as the reference receptor. The Licensee has demonstrated that the described methods and relevant parameters have followed the conservative approaches provided by NUREG-0133 and Regulatory Guide 1.109. However, the Licensee has not specifically included iodine-133 in Table 1.1-1 of the submittal. Also, the Licensee has not shown the derivation of the site-specific dose rate factors provided in the proposed Table 1.1-12.

Evaluation of the cumulative dose is to ensure that the quarterly and annual dose design objectives specified in RETS are not exceeded.

For liquid releases, the Licensee has identified fish consumption from the Connecticut River as the viable pathway. In the calculation, the Licensee has used the suggested values given in Regulatory Guide 1.109. As in the case of dose rate calculation, the Licensee has used the maximally exposed individual as the reference receptor. To correctly assess the cumulative dose, the Licensee intends to estimate the dose once per 31 days.

Evaluation of the cumulative dose from noble gas releases includes both beta and gamma and air doses at and beyond the site boundary. The critical organs under consideration are the total body and skin for gamma and beta radiation, respectively. Again, the Licensee has used the maximum (X/Q) values as discussed earlier and has followed the methodology and parameters of NUREG-0133 and Regulatory Guide 1.109. However, the Licensee should clearly define the average gamma dilution factor [X/Q^{γ}] and the method to calculate it. Also, the Licensee should include the derivation of the site-specific dose factors shown in the proposed Table 1.1-12.

For iodine-131, tritium, and particulates with half-lives greater than 8 days, the Licensee has provided a method to demonstrate that cumulative doses calculated from the release meet both quarterly and annual design objectives. Again, in the Licensee's Table 1.1-1, iodine-133 should specifically be included in the dose calculations.

Using the existing methodology for gaseous and liquid dose calculations, the Licensee has demonstrated a procedure to determine the monthly dose and to ensure that the design objectives for the liquid radwaste system and the ventilation exhaust system are not exceeded.

Adequate flow diagrams defining the effluent paths and components of the radioactive liquid and gaseous waste treatment systems have been provided by the Licensee. Radiation monitors specified in the Licensee-submitted RETS are also properly identified in the flow diagrams, except that the Licensee has not designated the hydrogen monitors. Also, the Licensee should delete the dose projection for the AOG treatment system, as the Licensee made a commitment to operate the AOG treatment system whenever the condenser air ejector system is in operation. The Licensee should also add a section for monthly dose projection for the ventilation exhaust treatment system.

The Licensee has provided a description of sampling locations in the ODCM and has identified them in Tables 4.1 and 4.2 and also in Figures 4-1 through 4-5 of that document. This description is consistent with the sampling locations specified in the Licensee's RETS Table 3.9.3 on environmental monitoring.

The Licensee has provided a method to assess the total dose (40CFR190 requirement) including the direct radiation, which satisfies the total dose provision of NUREG-0473.

In summary, except for the deficiencies discussed above, the Licensee's ODCM uses documented and approved methods that are consistent with the methodology and guidance in NUREG-0133, and therefore is an acceptable reference.

4. CONCLUSIONS

Table 1 summarizes the results of the final review and evaluation of the submittal for the Vermont Yankee Nuclear Power Station proposed Radiological Effluent Technical Specifications (RETS). The following conclusions have been reached:

1. The Licensee's proposed Radiological Effluent Technical Specifications (RETS) submitted January 23, 1984 [23] meet the intent of NUREG-0473, "Radiological Effluent Technical Specifications."

2. The Licensee's Offsite Dose Calculation Manual (ODCM) submitted July 14, 1983 [20] uses documented and approved methods that are consistent with the criteria of NUREG-0133 and applicable to the Vermont Yankee Nuclear Power Station, with the following exceptions:

   - The Licensee has not made a commitment to specifically include iodine-133 for the dose rate and dose calculations, as indicated by the Licensee's Table 1.1-1.

   - The Licensee has not clearly explained the derivation of the effective average gamma dilution factor $[X/Q^{\gamma}]$, and has not provided a method for its derivation.

   - The Licensee's service water monitor (Section 2.2.2 of the Licensee's submittal) does not have a specified LLD (see proposed RETS Table 4.8.1), and the setpoint is currently set at three times the background level. An alternative is needed to ensure that such a setpoint would correctly pick up a potential radioactivity leakage into the service water.

   - The Licensee has included a dose projection for the AOG treatment system, which is inconsistent with the RETS submittal and should therefore be deleted.

   - The Licensee has not included a dose projection for the ventilation exhaust treatment system as specified by the submittal.

   - The Licensee has not designated hydrogen monitors in Figure 6-2 of the ODCM submittal.

   - In the last line of page 1-6 of the Licensee's submittal, the number 2×10^{4} should read 2×10^{-4} to be consistent with the RETS submittal.

Table 1. Evaluation of Proposed Radiological Effluent Technical Specifications (RETS), Vermont Yakee Nuclear Power Station

RETS Requirement	Technical Specifications: NRC Staff Model RETS NUREG-0473 (Section)	Technical Specifications: Licensee Proposal (Section)	Replaces or Updates Existing Tech. Spec. (Section)	Evaluation
Effluent Instrumentation	3/4.3.3.10, 3/4.3.3.11	3.9.A, 3.9.B	3.9	Meets the intent of NRC criteria
Radioactive Effluent Concentrations	3/4.11.1.1, 3/4.11.2.1	3.8.A, 3.8.E	3.8	Meets the intent of NRC criteria
Offsite Doses	3/4.11.1.2, 3/4.11.2.2, 3/4.11.2.3, 3/4.11.4	3.8.B, 3.8.F, 3.8.G, 3.8.M	3.8	Meets the intent of NRC criteria
Effluent Treatment	3/4.11.1.3, 3/4.11.2.4, 3/4.11.2.5	3.8.C, 3.8.H, 3.8.I	Not addressed	Meets the intent of NRC criteria
Radioactivity Inventory Limits	3/4.11.1.4, 3/4.11.2.7	3.8.D, 3.8.K	3.8	Meets the intent of NRC criteria
Explosive Gas Mixtures	3/4.11.2.6	3.8.J	Not addressed	Meets the intent of NRC criteria
Solid Radioactive Waste	3/4.11.3	3.8.N	Not addressed	Meets the intent of NRC criteria
Environmental Monitoring	3/4.12.1	3.9.C	3.9.D	Meets the intent of NRC criteria
Audits and Reviews	6.5.1, 6.5.2	6.2.A, 6.2.B	6.5.2, 6.5.3	Meets the intent of NRC criteria
Procedures and Records	6.8, 6.10	6.5, 6.6	6.5, 6.6	Meets the intent of NRC criteria
Reports	6.9.1.11, 6.9.1.12, 6.9.2	6.7.C.1, 6.7.C.2, 6.7.C.3	6.7	Meets the intent of NRC criteria
Implementation of Major Programs	6.13, 6.14, 6.15	6.12, 6.13, 6.14	Not addressed	Meets the intent of NRC criteria

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3. Title 10, Code of Federal Regulations, Part 50, Appendix I, Section V, "Effective Dates"

4. Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation"

5. Title 40, Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

6. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

7. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

8. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 2
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12. W. Meinke (NRC)
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18. Franklin Research Center
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Vermont Yankee Nuclear Corporation
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NRC Docket No. 50-271

20. Vermont Yankee Offsite Dose Calculation Manual, Draft
Vermont Yankee Nuclear Corporation
July 14, 1983
NRC Docket No. 50-271

21. S. Pandey/S. Chen (FRC)
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Subject: FRC's Comments on Vermont Yankee's Draft RETS submittal
March 28, 1983

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Memo to W. Meinke (NRC)
Subject: FRC's Comments on Vermont Yankee's Draft ODCM submittal
August 18, 1983

23. L. H. Heider (VYNPC)
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Subject: Revised Vermont Yankee Radiological Effluent Technical Specifications (RETS)
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