Enclosure A

Revised Technical Specification Pages

1.0 DEFINITIONS

X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Y. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Vital Fire Suppression Water System - The vital fire suppression water system is that part of the fire suppression system which protects those instruments, components, and systems required to perform a safe shutdown of the reactor. The vital fire suppression system includes the water supply, pumps, and distribution piping with associated sectionalizing valves, which provide immediate coverage of the Reactor Building, Control Room Building, and Diesel Generator Rooms.

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, Revision 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. Deleted

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

GG. Deleted

HH. Deleted

BASES: 3.2 (Cont'd)

setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of nonsafety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip

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 dose limits for Members of the Public specified in 10CFR20.

Specification:

A. Liquid Effluents: Concentration

1. The concentration of radioactive material in liquid effluents released to Unrestricted Areas shall be limited to 10 times the concentrations specified in Appendix B to 10CFR Part 20.1001 - 20.2401, Table 2, Column 2 for radionuclides other than noble gases and 2×10^{-4} uCi/ml total activity concentration for all dissolved or entrained noble gases.

2. With the concentration of radioactive material in liquid effluents released to Unrestricted Areas 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.

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 dose limits for Members of the Public 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 to Unrestricted Areas are limited to the values in Specification 3.8.A.1.

3.8 LIMITING CONDITIONS FOR OPERATION	4.8 SURVEILLANCE REQUIREMENTS
B. Liquid Effluents: Dose	B. Liquid Effluents: Dose
1. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to Unrestricted Areas 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.	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.
C. Liquid Radwaste Treatment	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 effluents released to Unrestricted Areas, when averaged with all other liquid releases over the last month, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.	1. See Specification 4.8.B.1.

3.8 LIMITING CONDITIONS FOR OPERATION	4.8 SURVEILLANCE REQUIREMENTS
L. Primary Containment	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 of Iodine-131, Iodine-133 or radionuclides in particulate form with half-lives greater than 8 days exceed the levels specified in Appendix B to 10CFR20.1001 - 20.2401, Table 1, Column 3.	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.
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.	
M. Total Dose (40CFR190)	M. Total Dose
1. The dose or dose commitment to a member of the public* in areas at and beyond the Site Boundary 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.	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.

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

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 for Particulates	Prior to each release/ Each Purge	Principal Gamma Emitters[d,g] and I-131	1×10^{-9} (g)
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,g] and I-131	1×10^{-11}
	Continuous[e]	Once per month Composite Particulate Sample	Gross Alpha	1×10^{-11}
	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 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} µCi/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: (a) 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} µCi/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 Annual Radioactive 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, 3.3.F.1 and 3.8.G.1.

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.

g. Lower Limit of Detection (LLD) applies only to particulate form radionuclides identified in Table Notation d. above.

BASES:

3.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents: Concentration

This specification is provided to ensure that at any time the concentration of radioactive materials released in liquid waste effluents from the site above background (Unrestricted Area for liquids is at the point of discharge from the plant discharge into Connecticut River) will not exceed 10 times the concentration levels specified in 10CFR Part 20.1001-20.2401, Appendix B, Table 2, Column 2. These requirements provide operational flexibility, compatible with considerations of health and safety, which may temporarily result in releases higher than the absolute value of the concentration numbers in Appendix B, but still within the annual average limitation of the Regulation. Compliance with the design objective doses of Section II.A of Appendix I to 10CFR Part 50 assure that doses are maintained ALARA, and that annual concentration limits of Appendix B to 10CFR20.1001-20.2401 will not be exceeded.

The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radionuclide and that an effluent concentration in air (submersion dose equal to 500 mrem/yr) was converted to an equivalent concentration in water.

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, 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

BASES: 3.8 (Cont'd)

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 20.1001 - 20.2401, Appendix B, Table 2, Column 2, at the nearest potable water supply and in the nearest surface water supply in an Unrestricted Area.

E. Gaseous Effluents: Dose Rate

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. This instantaneous dose rate limit allows for operational flexibility when off normal occurrences may temporarily increase gaseous effluent release rates from the plant, while still providing controls to ensure that licensee meets the dose objectives of Appendix I to 10CFR50.

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 highest impacted cow.

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

BASES: 3.8 (Cont'd)

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.

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 (SBGT) so that the annual dose limits of 10CFR Part 20 for Members of the Public in areas at and beyond 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". The specification requires the use of SBGT only when Iodine-131, Iodine-133 or radionuclides in particulate form with half-lives greater than 8 days in containment exceeds the levels in Table 1, Column 3, to Appendix B of 10CFR 20.1001-20.2401 since the filter system is not considered effective in reducing noble gas radioactivity from gas streams.

M. Total Dose (40CFR190)

This specification is provided to meet the dose limitations of 40CFR Part 190 to Members of the Public in areas at and beyond the Site Boundary. 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

BASES: 3.8 (Cont'd)

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 20.2203(a)(4), 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. 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 subjects them to occupational exposures. For individuals in controlled areas who are considered Members of the Public per 10CFR20, the dose limits of 10CFR20.1301 apply since the licensee has the authority to control and limit access to these areas.

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.

TABLE 3.9.1 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 3.8.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 4.9.1 NOTATION

(1) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Institute for Standards and Technology) 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 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 Institute for Standards and Technology) 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.

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 10 times the concentration limits of Appendix B to 10CFR20.1001-20.2401, Table 2, Column 2, values.

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 above 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 design bases dose rates identified in 3.8.E.1. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

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

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, boundary for liquid effluents, as well as areas defined per 10CFR20 as "controlled areas" and "unrestricted areas" are 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) or hafnium, or a combination of the two.

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.

5.5 Spent and New Fuel Storage

A. The new fuel storage facility shall be such that the effective multiplication factor (K_{eff}) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

B. The K_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

C. Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

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 20.1601, "Control of Access to High Radiation Areas." In lieu of the "control device" or "alarm signal" required by Paragraph 20.1601(a), each high radiation area in which the intensity of radiation is greater than 100 mrem/hr at 30 cm, but less than 1000 mrem/hr at 30 cm, 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 at 30 cm, but less than 500 rad/hr at 1 meter. 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 Radiation Protection Manager.

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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.

6.6 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years:

1. Records of normal plant operation, including power levels and periods of operation at each power level.

2. Records of principal maintenance activities, including inspection and repair or principal items of equipment pertaining to nuclear safety.

3. Records of reportable occurrences.

4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements are being met.

5. Records of any special reactor test or experiments.

6. Records of changes made in the Operating Procedures.

7. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

8. Results of annual physical inventory verifying accountability of licensed sources on record.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Records of substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Records of changes and drawing changes made to the plant as it is described in the Safety Analysis Report.

3. Records of plant radiation and contamination surveys.

4. Records of new and spent fuel inventory, transfers of fuel, and assembly histories.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant for whom monitoring was required in accordance with 10 CFR 20.

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.

12. Records of radioactive shipments.

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.

1/ This tabulation supplements the requirements of 20.2206 of 10CFR Part 20.

The dose assignment to various duty functions may be estimates based on Self-Reading Dosimeter (SRD), 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 Statistical 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.

4. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: (a) The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a, (b) The K_f core flow adjustment factor for Specification 3.11.C., (c) The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a, (d) The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a, 2.1.B.1, and 3.11.B, and (e) The Power/Flow Exclusion Region for Specifications 3.6.J.1a and 3.6.J.1b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).

Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).

(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 20.2203(a)(4), 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.

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 for the purpose of demonstrating compliance with 10CFR50, Appendix I, 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 Annual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s) and

b. A determination that the change will maintain the level of radioactive effluent control required by 10CFR20.1302, 40CFR190, 10CFR50.36a, and Appendix I to 10CFR Part 50, and not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

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

3. Shall be submitted to the Commission in the form of a copy of the affected pages of the ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the magin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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 Annual 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;

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

Enclosure B

Marked-Up Technical Specification Pages

1.0 DEFINITIONS

X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Y. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Vital Fire Suppression Water System - The vital fire suppression water system is that part of the fire suppression system which protects those instruments, components, and systems required to perform a safe shutdown of the reactor. The vital fire suppression system includes the water supply, pumps, and distribution piping with associated sectionalizing valves, which provide immediate coverage of the Reactor Building, Control Room Building, and Diesel Generator Rooms.

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, Revision 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.

DELETED

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. Deleted

HH. Deleted

BASES: 3.2 (Cont'd)

setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of nonsafety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident, and 10CFR20 limits are not exceeded for gross fuel failure during reactor operations. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

DELETE

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip

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. for Members of the Public

Specification:

A. Liquid Effluents: Concentration

1. The concentration of radioactive material in liquid effluents released from the site to Unrestricted Areas shall be limited to the concentrations specified in 10CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases and 2×10^{-4} uCi/ml total activity concentration for all dissolved or entrained noble gases. Appendix B to 10CFR Part 20.1001-20.2401, Table 2, Column 2 10 Times

2. With the concentration of radioactive material in liquid effluents released from the site to Unrestricted Areas 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.

4.8 SURVEILLANCE REQUIREMENTS

4.8 RADIOACTIVE EFFLUEN

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. for Members of the Public

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 th ODCM to assure that the concentrations at the point of release to Unrestricted Areas are limited to the values in Specification 3.8.A.1.

3.8 LIMITING CONDITIONS FOR OPERATION	4.8 SURVEILLANCE REQUIREMENTS
B. Liquid Effluents: Dose	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~~ to Unrestricted Areas shall be limited to the following:	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.
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.	
C. Liquid Radwaste Treatment	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~~ released to Unrestricted Areas, when averaged with all other liquid releases over the last month, would exceed 0.06 mrem to the total body, or 0.2 mrem to any organ.	1. See Specification 4.8.B.1.

3.8 LIMITING CONDITIONS FOR OPERATION	4.8 SURVEILLANCE REQUIREMENTS
L. Primary Containment	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.	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.
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.	
M. Total Dose (40CFR190)	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.	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.

of Iodine-131, Iodine-133 or radionuclides in particulate form with half-lives greater than 8 days exceed the levels specified in Appendix B to 10CFR20.1001-20.2401, Table 1, Column 3.

in areas at and beyond the Site Boundary

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

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] (g) and I-131	1×10^{-4} (g)
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,g]	1×10^{-11}
	Continuous[e]	Once per month Composite Particulate Sample	Gross Alpha	1×10^{-11}
	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}

/ Each Purge Grab Sample for Particulates

and I-131

TABLE 4.8.2 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} µCi/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: (a) 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} µCi/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 Annual Radioactive 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.

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.

g. Lower Limit of Detection (LLD) applies only to particulate form radionuclides identified in Table Notation d. above.

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 exposure 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, 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 197..

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

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not exceed 10 times the concentration levels specified in 10CFR Part 20.1001-20.2401, Appendix B, Table 2, Column 2. These requirements provide operational flexibility, compatible with considerations of health and safety, which may temporarily result in releases higher than the absolute value of the concentration numbers in Appendix B, but still within the annual average limitation of the Regulation. Compliance with the design objective doses of Section II.A of Appendix I to 10CFR Part 50 assure that doses are maintained ALARA, and that annual concentration limits of Appendix B to 10CFR20.1001-20.2401 will not be exceeded.

BASES: 3.8 (Cont'd)

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.

10CFR Part 20.1001-20.2401, Appendix B, Table 2, Column 2

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.

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 [10 CFR 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.

This instantaneous dose rate limit allows for operational flexibility when off normal occurrences may temporarily increase gaseous effluent release rates from the plant, while still providing controls to ensure that licensee meets the dose objectives of Appendix I to 10CFR50.

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~~ highest impacted cow.

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

BASES: 3.8 (Cont'd)

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.

L. Primary Containment (MARK I) (SBGT)

for Members of the Public in areas and beyond

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".

Insert B

M. Total Dose (40CFR190)

to Members of the Public in areas at and beyond the Site Boundary

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

Page 188 Insert:

B

The specification requires the use of SBGT only when Iodine-131, Iodine-133 or radionuclides in particulate form with half-lives greater than 8 days in containment exceeds the levels in Table 1, Column 3, to Appendix B of 10CFR20.1001-20.2401 since the filter system is not considered effective in reducing noble gas radioactivity from gas streams.

BASES: 3.8 (Cont'd)

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.

subjects them to occupational exposures. For individuals in controlled areas who are considered Members of the Public per 10CFR20, the dose limits of 10CFR20.1301 apply since the licensee has the authority to control and limit access to these areas.

TABLE 3.9.1 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 3.8.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 4.9.1 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 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~~ Institute for Standards and Technology) 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.

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~~. 10 times the concentration limits of Appendix B to 10CFR20.1001-20.2401, Table 2, Column 2, Values. 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~~ above 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~~ design bases dose rates identified in 3.8.E.1. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

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

5.0 DESIGN FEATURES

as well as areas defined per 10CFR20 as "controlled areas" and "unrestricted areas" are

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) or hafnium, or a combination of the two.

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.

5.5 Spent and New Fuel Storage

A. The new fuel storage facility shall be such that the effective multiplication factor (K_{eff}) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

B. The K_{eff} of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

C. Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

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.

20.1601, "Control of Access to High Radiation Areas."

greater than 100 mrem/hr at 30 cm, but less than 1000 mrem/hr at 30 cm

20.1601(a)

at 30 cm, but less than 500 rad/hr at 1 meter

Radiation Protection Manager

* 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.

6.6 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years:

1. Records of normal plant operation, including power levels and periods of operation at each power level.

2. Records of principal maintenance activities, including inspection and repair or principal items of equipment pertaining to nuclear safety.

3. Records of reportable occurrences.

4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements are being met.

5. Records of any special reactor test or experiments.

6. Records of changes made in the Operating Procedures.

~~7. Records of radioactive shipments.~~

7 8. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

8 9. Results of annual physical inventory verifying accountability of licensed sources on record.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Records of substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Records of changes and drawing changes made to the plant as it is described in the Safety Analysis Report.

3. Records of plant radiation and contamination surveys.

4. Records of new and spent fuel inventory, transfers of fuel, and assembly histories.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

for whom monitoring was required

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.

12. Records of radioactive shipments

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.

1/ This tabulation supplements the requirements of 20.2206 of 10CFR Part 20.

Self-Reading (SRD)

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 Statistical 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.

4. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following: (a) The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a, (b) The K_f core flow adjustment factor for Specification 3.11.C., (c) The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a, (d) The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a, 2.1.B.1, and 3.11.B, and (e) The Power/Flow Exclusion Region for Specifications 3.6.J.1a and 3.6.J.1b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

Report, D. M. VerPlanck, "Methods for the Analysis of Boiling Water Reactors Steady State Core Physics," YAEC-1238, March 1981 (Approved by NRC, SER, dated September 15, 1982).

Report, J. M. Holzer, "Methods for the Analysis of Boiling Water Reactors Transient Core Physics," YAEC-1239P, August 1981 (Approved by NRC SER, dated September 15, 1982).

Report, S. P. Schultz and K. E. St.John, "Methods for the Analysis of Guide Fuel Rod Steady-State Thermal Effects (FROSSTEY): Code/Model Description Manual," YAEC-1249P, April 1981 (Approved by NRC SER, dated September 27, 1985).

(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.

20.2203(a)(4)

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.

for the purpose of demonstrating compliance with 10CFR50, Appendix I,

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 Annual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

maintain the level of radioactive effluent control required by 10CFR20.1302, 40CFR190, 10CFR50.36a, and Appendix I to 10CFR Part 50, and not adversely impact the accuracy or reliability of effluent dose, or setpoint calculations.

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 which were changed with each page numbered and provided with the revision number, together with appropriate analyses or evaluations justifying the change(s); and

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).

Insert C

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 Annual 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;

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

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C

3. Shall be submitted to the Commission in the form of a copy of the affected pages of the ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g. month/year) the change was implemented.