August 30, 1985

Docket No. 50-293

Mr. William D. Harrington
Senior Vice President, Nuclear
Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Dear Mr. Harrington:

The Commission has issued the enclosed Amendment No. 89 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated May 14, 1985. These changes incorporate revised radiological effluent and environmental monitoring limiting conditions for operation, action statements, and surveillance requirements.

To facilitate implementation of the Pilgrim radiological effluent Technical Specifications, please note that your Technical Specifications for explosive gas mixtures (3.8.F.1 and Table 4.8-4) have been accepted on an interim basis until a formal policy on explosive gas monitoring has been established by the NRC.

A copy of the related Safety Evaluation is also enclosed.

Further, we have reviewed your "Offsite Dose Calculation Manual" (ODCM) submitted June 16, 1983 and find it uses documented and approved methods that are consistent with the methodology and guidelines in NUREG-0133 and, therefore, is an acceptable reference.

Sincerely,

Original signed by/

Paul H. Leech, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 89 to License No. DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. William D. Harrington
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Pilgrim Nuclear Power Station

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
License No. DPR-35

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

   A. The applications for amendment by Boston Edison Company (the licensee) dated May 14, 1985, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 89 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of March 1, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 30, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 89

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the areas of change.

Pages
ii
iii
5a
5b new page
43
44
56
64
72
76
77
177 through 193
193a through 193f - new pages
223
223a
229 through 249 - new pages

1.0 DEFINITIONS (Continued)

U. 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 total maximum combined interval time for any three consecutive tests shall not exceed 3.25 times the specified interval. The operating cycle interval is considered to be 18 months and the tolerances stated above are applicable.

V. 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 may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable.

W. Fire Suppression Water System - A fire suppression water system shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include hydrant post indicator valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

X. Staggered Test Basis - A staggered test basis shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

Y. Source Check - A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

Z. Offsite Dose Calculation Manual (ODCM) - An offsite dose calculation manual (ODCM) shall be a manual containing the current methodology and parameters to be used for the calculation of offsite doses due to radioactive gaseous and liquid effluents, the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints, and the conduct of the Radiological Environmental Monitoring Program.

1.0 DEFINITIONS (Continued)

AA. Action - Action shall be that part of a specification which prescribes remedial measures required under designated conditions.

BB. Member(s) of the Public[1] - Member(s) 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 persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the site.

CC. Site Boundary[1] - The site boundary is shown in Figure 1.6-1 in the FSAR.

DD. Radwaste Treatment System

1. Gaseous Radwaste Treatment System - The gaseous radwaste treatment system is that system identified in Figure 4.8-2.

2. Liquid Radwaste Treatment System - The liquid radwaste treatment system is that system identified in Figure 4.8-1.

[1] See FSAR Figure 1.6-1

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
C. Control Rod Block Actuation	C. Control Rod Block Actuation
1. The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.C.	1. Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.C. System logic shall be functionally tested as indicated in Table 4.2.C.
2. The minimum number of operable instrument channels specified in Table 3.2.C for the Rod Block Monitor may be reduced by one in one of the trip systems for maintenance and/or testing, provided that this condition does not last longer than 24 hours in any thirty day period.	
D. Radiation Monitoring Systems - Isolation & Initiation Functions	D. Radiation Monitoring Systems - Isolation & Initiation Functions
1. Reactor Building Isolation and Control System and Standby Gas Treatment System The limiting conditions for operation are given in Table 3.2.D.	1. Reactor Building Isolation and Control System and Standby Gas Treatment System Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.D. System logic shall be functionally tested as indicated in Table 4.2.D.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
E. Drywell Leak Detection The limiting conditions of operation for the instrumentation that monitors drywell leak detection are given in Table 3.2.E.	E. Drywell Leak Detection Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2.E.
F. Surveillance Information Readouts The limiting conditions for the instrumentation that provides surveillance information readouts are given in Table 3.2.F.	F. Surveillance Information Readouts Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

PNPS

TABLE 3.2.D

RADIATION MONITORING SYSTEMS THAT INITIATE AND/OR ISOLATE

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Action (2)
2	Refuel Area Exhaust Monitors	Upscale, <100 mr/hr	A or B
2	Refuel Area Exhaust Monitors	Downscale	A or B

NOTES FOR TABLE 3.2.D

1. Whenever the systems are required to be operable, there shall be two operable or tripped trip systems. If this cannot be met, the indicated action shall be taken.

1. Action

   A. Cease operation of the refueling equipment.
   B. Isolate secondary containment and start the standby gas treatment system.

PNPS
TABLE 4.2.D
MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS

	Instrument Channels	Instrument Functional Test	Calibration	Instrument Check (2)
1)	Refuel Area Exhaust Monitors - Upscale	(1)	Once/3 months	Once/day
2)	Refuel Area Exhaust Monitors - Downscale	(1)	Once/3 months	Once/day

Logic System Functional Test (4) (6)	Frequency
1) Reactor Building Isolation	Once/6 months
2) Standby Gas Treatment System Actuation	Once/6 months

HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Four radiation monitors are provided which initiate the Reactor Building Isolation and Control System and operation of the standby gas treatment system. The instrument channels monitor the radiation from the refueling area ventilation exhaust ducts.

Four instrument channels are arranged in a 1 out of 2 twice trip logic.

Trip settings of < 100 mr/hr for the monitors in the refueling area ventilation exhaust ducts are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the Reactor Building via the normal ventilation path but rather all the activity is processed by the standby gas treatment system.

Flow integrators are used to record the integrated flow of liquid from the drywell sumps. The alarm unit in each integrator is set to annunciate before the values specified in Specification 3.6.C are exceeded. A system whereby the time interval to fill a known volume will be utilized to provide a back-up to the flow integrators. An air sampling system is also provided to detect leakage inside the primary containment.

is shown by Curve No. 2. Note that the unavailability is lower as expected for a redundant system and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more unusual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in Curve No. 3. Note that the minimum occurs at about 40,000 hours, much longer than for cases 1 and 2. Also, the minimum is not nearly as low as Case 2 which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. more than one channel should not be bypassed for testing at any one time.

The radiation monitors in the refueling area ventilation duct which initiate building isolation and standby gas treatment operation are arranged in two 1 out of 2 logic systems. The bases given above for the rod blocks apply here also and were used to arrive at the functional testing frequency. Based on experience with instruments of similar design, a testing interval of once every three months has been found adequate.

4.2 BASES (Cont'd)

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the recirculation pump trip and alternate rod insertion systems incorporate analog transmitters and are a new, improved line of BWR instrumentation. The calibration frequency is once per operating cycle which is consistent with both the equipment capabilities and the requirements for similar equipment used by other reactor vendors. The calibration frequency of the trip units is proposed to be quarterly, the same as other similar protective instrumentation. Likewise, the test frequency is specified at monthly like that of other protective instrumentation. A sensor check is proposed once per day; this is considered to be an appropriate frequency, commensurate with the design applications and the fact that the recirculation pump trip and alternate rod insertion systems are backups to existing protective instrumentation.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8 RADIOACTIVE EFFLUENTS	4.8 RADIOACTIVE EFFLUENTS
A. Liquid Effluents Concentration	A. Liquid Effluents Concentration
Applicability: At all times.	Specification: 1. The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.8-1. 2. The results of pre-release analyses shall be used with calculational methods in the Offsite Dose Calculation Manual (ODCM) to assure that the concentration at the point of release is limited to the values in Specification 3.8.A.1.
Specification: 1. The concentration of radioactive material released at any time from the site to areas at and beyond the site boundary shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration of individual isotopes shall be limited to 2 x 10^{-4} Ci/ml. Action With the concentration of radioactive material released from the site to areas at and beyond the site boundary exceeding the above limits, without delay restore concentration within the above limits.	
B. Radioactive Liquid Effluent Instrumentation	B. Radioactive Liquid Effluent Instrumentation
Applicability: As shown in Table 3.8-1.	Specification: 1. The setpoints for monitoring instrumentation shall be determined in accordance with the ODCM. 2. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable at the frequencies shown in Table 4.8-2.
Specification: 1. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.8-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.A.1 are not exceeded	

3.8.B Radioactive Liquid Effluent Instrumentation (Continued)

during periods when liquid wastes are being discharged via the radwaste discharge header.

For releases other than the radwaste discharge header, the above specification does not apply, these releases shall be made in accordance with Action 1 of Table 3.8-1.

Action

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.8.A.1 are met, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or change the setpoint so that it is acceptably conservative or declare the channel inoperable.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the action shown in Table 3.8-1.

C. Liquid Radwaste Treatment

Applicability:

At all times.

Specification:

1. The liquid radwaste treatment system shall be maintained and used to reduce the radioactive materials in liquid wastes

C. Liquid Radwaste Treatment

Specification:

1. Doses due to liquid releases at and beyond the site boundary shall be calculated at least once per 31-day period in accordance with the ODCM, only if releases in that period have occurred.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.C. Liquid Radwaste Treatment (Continued) prior to their discharge when the dose due to liquid effluent releases to areas at and beyond the site boundary averaged over a 31-day period would exceed 0.06 mrem to the total body or 0.20 mrem to any organ. Action a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days a special report which includes the following information: 1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability. 2. Action(s) taken to restore the inoperable equipment to operable status. 3. Summary description of action(s) taken to prevent a recurrence.	4.8.C. Liquid Radwaste Treatment (Continued) 2. The liquid radwaste treatment system schematic is shown in Figure 4.8-1.
D. Gaseous Effluents Dose Rate Applicability: At all times. Specification: 1. The instantaneous dose rate due to radioactive materials released in gaseous effluents	D. Gaseous Effluents Dose Rate Specification: 1. The instantaneous dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.8.D.1.a on a continuous basis using the noble gas activity monitors

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.D. Gaseous Effluents Dose Rate (Continued) from the site to areas at and beyond the site boundary (see FSAR Figure 1.6-1) shall be limited to the following: a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ. Action With the instantaneous dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).	4.8.D. Gaseous Effluents Dose Rate (Continued) with appropriate setpoints and in accordance with the ODCM. 2. The instantaneous dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.8.D.1.b in accordance with the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.8-3.
E. Radioactive Gaseous Effluent Instrumentation Applicability: As shown in Table 3.8-2. Specification: 1. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.8-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.D.1 are not exceeded.	E. Radioactive Gaseous Effluent Instrumentation Specification: 1. The setpoints shall be determined in accordance with ODCM. 2. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated operable at the frequencies shown in Table 4.8-4.

LIMITING CONDITIONS FOR OPERATION

3.8.E. Radioactive Gaseous Effluent Instrumentation (Continued)

Action

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.8.D.1 are met, change the setpoint so that it is acceptably conservative or declare the channel inoperable.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the action shown in Table 3.8-2.

F. Gaseous Effluent Treatment

Applicability:

The augmented offgas system shall be put into service prior to reaching 50 percent reactor power during startup.

Action

a. With gaseous effluents being discharged for more than 14 days without treatment, prepare and submit to the Commission within 30 days, a special report which includes the following information:

1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability.

2. Action(s) taken to restore the inoperable equipment to operable status.

SURVEILLANCE REQUIREMENTS

F. Gaseous Effluent Treatment

Specification:

1. Augmented offgas annunciator operability shall be verified once per 12 hours.

2. The concentration of hydrogen in the augmented offgas treatment system shall be determined to be within the limits of Specification 3.8.F.1 by continuously monitoring the waste gases in the augmented offgas treatment system with the hydrogen monitor which is required to be operable by Table 3.8-2.

3.8.F. Gaseous Effluent Treatment (Continued)

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

Specification:

1. The concentration of hydrogen in the augmented offgas treatment system shall be limited to less than or equal to 2 percent by volume at the outlet of the augmented offgas recombiner. See also Action 5 for Item 4.a on Table 3.8-2.

Action

a. With the concentration of hydrogen in the augmented offgas treatment system greater than 2 percent by volume but less than or equal to 4 percent by volume, restore the concentration of hydrogen to within the limit within 48 hours or be in a cold shutdown condition within 24 hours.

G. Main Condenser

Applicability:

At all times when steam is available to the air ejectors.

Specification:

1. The gross radioactivity (beta and/or gamma) release rate of noble gases measured at the steam jet air ejector shall be limited to 500,000 μCi/sec (referenced to a 30-minute holdup).

Action

With the gross radioactivity (beta and/or gamma) release

G. Main Condenser

Specification:

1. The gross radioactivity (beta and/or gamma) release rate of noble gases from the steam jet air ejector shall be determined to be within the limit of Specification 3.8.G.1 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge of the steam jet air ejector (prior to dilution and/or discharge):

a. At least once per 31 days.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.8.G. Main Condenser (Continued) rate of noble gases at the steam jet air ejector exceeding 500,000 μCi/sec (referenced to a 30-minute holdup), restore the gross radioactivity release rate to within the limit within 72 hours or be in at least hot standby within the next 12 hours. See also Action 1 for Item 3.a on Table 3.8-2.	4.8.G. Main Condenser (Continued) b. When the average daily gross radioactivity release rate increases by 50 percent over the previous day, after factoring out increases due to changes in reactor thermal power level.
H. Mechanical Vacuum Pump Specification: 1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines whenever the main steam isolation valves are open. 2. If the limits of Specification 3.8.H.1 are not met, the vacuum pump shall be isolated.	H. Mechanical Vacuum Pump Specification: 1. At least once during each operating cycle verify automatic securing and isolation of the mechanical vacuum pump.

TABLE 3.8-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Action[2]
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	1	During actual discharge of liquid wastes	1
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	1	During actual discharge of liquid wastes	2
b. Discharge Canal[1]	NA	During actual discharge of liquid wastes	3

[1] Flow will be estimated based on the design flow rate of the operating circulating water pumps and/or the operating salt service water pumps.

[2] ACTION 1 With the number of operable channels less than required by the minimum channels operable requirement, effluent releases may be resumed 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. An independent verification of the release rate calculations is performed, and

c. An independent verification of the discharge valving is performed.

ACTION 2 With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that the flow rate is verified at least once per 4 hours during actual releases.

ACTION 3 Suspend all radioactive liquid effluent discharges if no dilution water is available.

TABLE 3.8-2
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Applicability	Parameter	Action[4]
1. Main Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor - Providing Alarm	1	'	Radioactivity Rate Measurement	3
b. Iodine Sampler Cartridge	1	'	Collect Halogen Sample	4
c. Particulate Sampler Filter	1	'	Collect Particulate Sample	4
d. Effluent System Flow Rate Measuring Device	1	'	System Flow Rate Measurement	2
e. Sampler Flow Rate Measuring Device	1	'	Sampler Flow Rate Measurement	2
2. Reactor Building Ventilation Effluent Monitoring System				
a. Noble Gas Activity Monitor - Providing Alarm	1	'	Radioactivity Rate Measurement	3
b. Iodine Sampler Cartridge	1	'	Collect Halogen Sample	4
c. Particulate Sampler Filter	1	'	Collect Particulate Sample	4

TABLE 3.8-2 (Continued)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	Instrument	Minimum Channels Operable	Applicability	Parameter	Action[4]
2.	Reactor Building Ventilation Effluent Monitoring System (Continued)				
	d. Effluent System Flow Rate Measurement Device	1	[1]	System Flow Rate Measurement	2
	e. Sampler Flow Rate Measurement Device	1	[1]	Sampler Flow Rate Measurement	2
3.	Steam Jet Air Ejector Radioactivity Monitor				
	a. Noble Gas Activity Monitor (Providing alarm and auto-isolation of stack)	1	[3]	Noble Gas Radio-activity Rate Measurement	1
4.	Augmented Offgas Treatment System Explosive Gas Monitoring				
	a. Hydrogen Monitor	1	[2]	Hydrogen Concentra-tion Measurement	5

TABLE 3.8-2 (Continued)
TABLE NOTATION

[1] During releases via this pathway.

[2] During augmented offgas treatment system operation.

[3] During operation of the steam jet air ejector.

[4] ACTION 1 With the number of operable channels less than required by the minimum channels operable requirement, gases from the steam jet air ejector may be released to the offgas system for up to 72 hours provided:

a. The augmented offgas treatment system is not bypassed, and

b. The offgas holdup system noble gas activity effluent monitor (downstream) is operable.

Otherwise, be in at least hot standby within 12 hours.

ACTION 2* With the number of operable channels 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.

ACTION 3* With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for activity within 24 hours.

ACTION 4* With the number of operable channels less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.8-3.

ACTION 5 With the number of operable channels less than required by the minimum channels operable requirement, operation of the augmented offgas holdup system may continue provided grab samples are collected at least once per 24 hours, analyzed within the following 4 hours, and the proper function of the recombiner is assured by monitoring recombiner temperature.

*Note: (For Actions 2, 3, and 4) If the instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

TABLE 4.8-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (Ci/ml)[a]
A. Batch Waste Release Tanks[c]	Each Batch	Prior to Release Each Batch	Principal Gamma Emitters[d]	5×10^{-7}
1. Non-treatable Releases (e.g., Neutralizer Sumps) and 2. Treatable Releases (e.g., Radwaste Tanks)			I-131	1×10^{-6}
			Dissolved and Entrained Gases	1×10^{-5}
	Composite from Each Batch	Monthly Composite[b]	H-3	1×10^{-5}
			Gross alpha	1×10^{-7}
	Composite from Each Batch	Quarterly Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases 1. Salt Service Water	Weekly grab sample	Weekly	Principal Gamma Emitters	5×10^{-7}

TABLE 4.8-1 (Continued)
TABLE NOTATION

(a) Refer to ODCM for LLD definition.

(b) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

(c) A batch release is the discharge of liquid wastes of a discrete volume.

(d) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall be analyzed and reported in the Semiannual Radioactive Effluent Release Report.

TABLE 4.8-2

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Channel Calibration	Channel Functional Test
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Liquid Radwaste Effluents Line	[1]	NA	Once per operating cycle [2]	Quarterly
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	[1]	NA	Once per operating cycle	Quarterly

[1] During or prior to release via this pathway.

[2] Previously established calibration procedures will be used for these requirements.

TABLE 4.8-3
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	(LLD)[a] (Ci/ml)
Main Stack and Rx Bldg. Vent	Monthly Grab Sample	Monthly	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[d]	Weekly Charcoal[c] Sample	I-131	1×10^{-12}
	Continuous[d]	Weekly Particulate[c] Sample	Principal Gamma Emitters[b] (I-131, others)	1×10^{-11}
	Continuous[d]	Monthly Composite Particulate Sample	gross alpha	1×10^{-11}
	Continuous[d]	Quarterly Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[d]	Continuous Noble Gas Monitor	Noble Gases Gross Gamma	1×10^{-6}

TABLE 4.8-3 (Continued)
TABLE NOTATION

(a) Refer to ODCM for LLD definition.

(b) The principal gamma emitters for which the LLD specification applies exclusively are 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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall be analyzed and reported in the Semiannual Radioactive Effluent Release Report.

(c) When the average daily gross radioactivity release rate increases by 50 percent over the previous day (after factoring out power level changes), the iodine and particulate filters shall be analyzed to determine the release rate for iodines and particulates.

(d) 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 Specification 3.8.D.

TABLE 4.8-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
1. Main Stack Effluent Monitoring System				
a. Noble Gas Activity Monitor (Two channels)	Daily[1]	Monthly	Once per operating cycle[4]	Quarterly
b. Iodine Sampler Cartridge	NA	NA	NA	NA
c. Particulate Sampler Filter	NA	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	Daily[1]	NA	Once per operating cycle	Quarterly
e. Sampler Flow Rate Measuring Device	Daily[1]	NA	Once per operating cycle	Quarterly
2. Reactor Building Ventilation Effluent Monitoring System				
a. Noble Gas Activity Monitor	Daily[1]	Monthly	Once per operating cycle[4]	Quarterly
b. Iodine Sampler Cartridge	NA	NA	NA	NA
c. Particulate Sampler Filter	NA	NA	NA	NA
d. Effluent System Flow Rate Measuring Device	Daily[1]	NA	Once per operating cycle	Quarterly

TABLE 4.8-4 (Continued)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Check	Source Check	Instrument Calibration	Instrument Functional Test
e. Sampler Flow Rate Measuring Device	Daily[1]	NA	Once per operating cycle	Quarterly
3. Steam Jet Air Ejector Radioactivity Monitor				
a. Noble Gas Activity Monitor	Daily[3]	NA	Once per operating cycle[4]	Quarterly
4. Augmented Offgas Treatment System Explosive Gas Monitoring System				
a. Hydrogen Monitor	Daily[2]	NA	Quarterly[5]	Monthly

[1]During releases via this pathway.

[2]During augmented offgas treatment system operation.

[3]During operation of the steam jet air ejector.

[4]Previously established calibration procedures will be used for these requirements.

[5]Calibrate at 2 points with standard gas samples differing by at least 1% but not exceeding 4%.

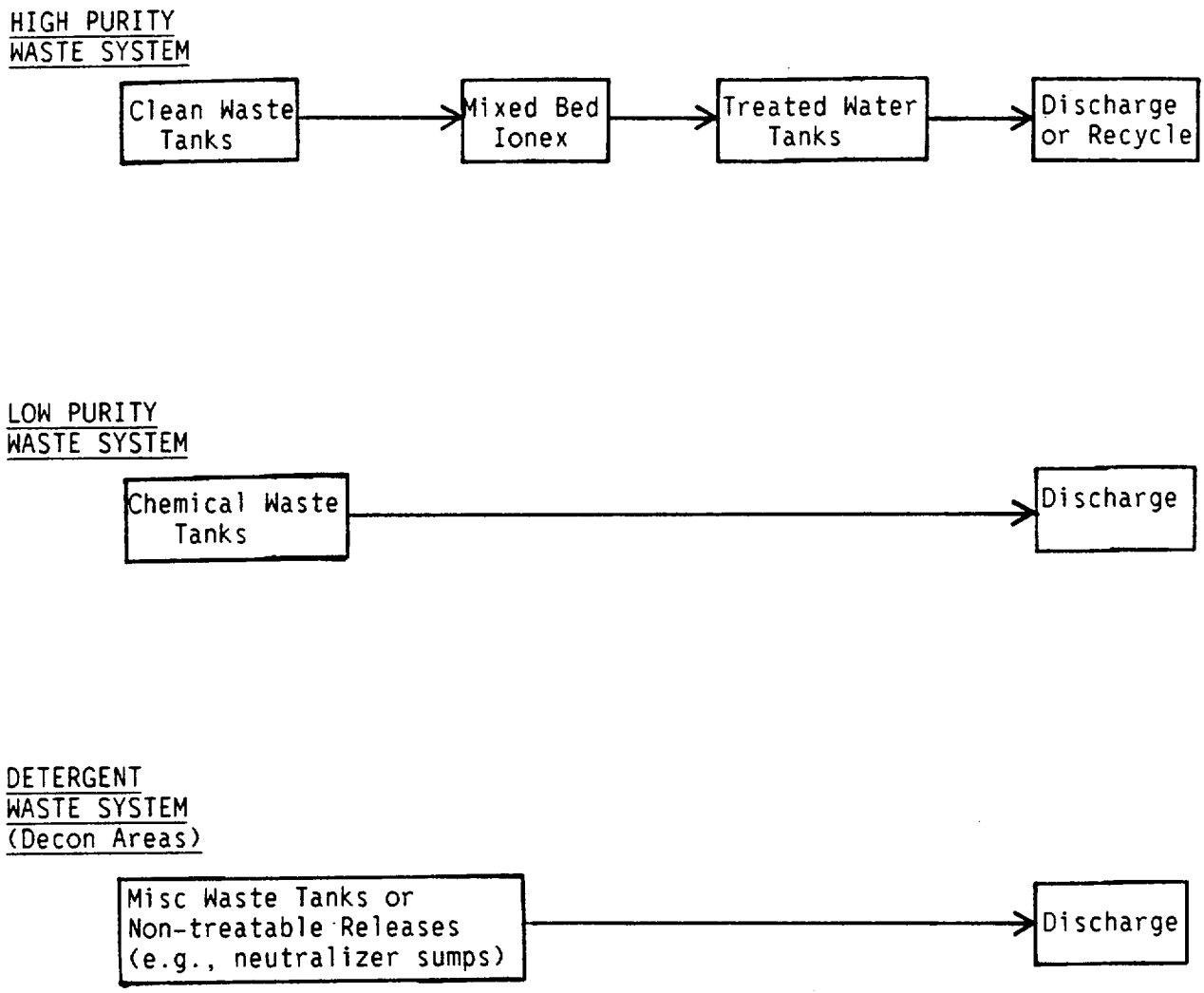
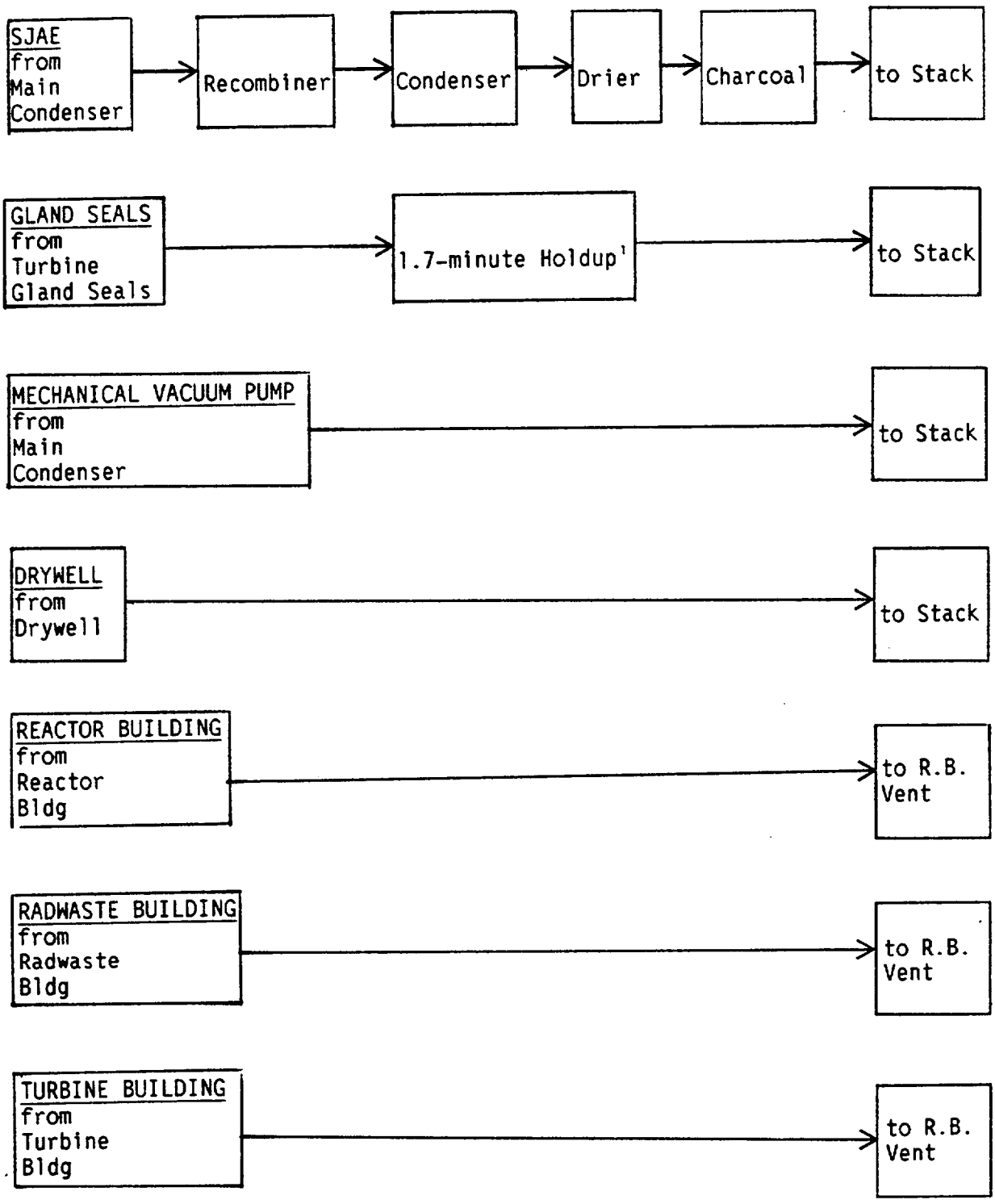


Figure 4.8-1 Liquid Radwaste Treatment System Schematic



[1] No significant effect in reducing offsite doses when compared to transit time required for releases to reach site boundary.

Figure 4.8-2 Gaseous Effluent Treatment Schematic

BASES

3/4.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents at and beyond the site boundary will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water at and beyond the site boundary will not result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a member of the public and (2) the limits of 10 CFR Part 20.106(e) to the population.

B. Radioactive 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/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the Offsite Dose Calculation Manual (ODCM) to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

C. Liquid Effluent Treatment

The requirement that the appropriate portions of this system 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 10 CFR Part 50.36a, General Design Criteria 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

D. Gaseous Effluents Dose Rate

This specification is provided to ensure that the dose rate at anytime at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public either within or outside the site boundary to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20.106(b). For members of the public who may at times be within the site boundary, the occupancy of the individual will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding

3/4.8.D Gaseous Effluents Dose Rate (Continued)

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 ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, 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.

E. Radioactive Gaseous Process and Effluent Monitoring 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 shall be calculated in accordance with NRC approved methods in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The process monitoring instrumentation includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the main condenser offgas treatment system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

F. Gaseous Effluent Treatment

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

Maintaining the concentration of hydrogen below it's flammability limits provides assurance that releases of radioactive materials will be controlled in conformance with the requirements of General Design Criteria 60 of Appendix A to 10 CFR Part 50.

G. Main Condenser

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to a member of the public at and beyond the site boundary will not exceed a small fraction of the limits of 10 CFR 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 10 CFR Part 50.

3/4.8.G Main Condenser (Continued)

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip. There is a fifteen minute delay before the air ejector off-gas isolation valve is closed. This delay is accounted for by the 30-minute holdup time of the off-gas before it is released to the stack.

Both instruments are required for trip but the instruments are so designed that any instrument failure gives a downscale trip. The trip settings of the instruments are set so that the instantaneous stack release rate limit given in Specification 3.8 is not exceeded.

H. Mechanical Vacuum Pump

The purpose of isolating the mechanical vacuum pump line is to limit the release of activity from the main condenser. During an accident, fission products would be transported from the reactor through the main steam lines to the condenser. The fission product radioactivity would be sensed by the main steam line radioactivity monitors, which initiate isolation.

6.9.C. Unique Reporting Requirements

1. Radioactive Effluent Release Report

A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released at and beyond the site boundary in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Appendix B of Regulatory Guide 1.21 (Revision 1) dated June, 1974.

2. Annual Radiological Environmental Monitoring Report

A report on the radiological environmental surveillance program for the previous calendar year of operation shall be submitted to the Director of the NRC Regional Office with a copy to the Director, Office of Nuclear Reactor Regulation as a separate document prior to May 1 of the year. The reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, operational controls and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of any land use surveys which affect the choice of sample locations. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

The Annual Radiological Environmental Monitoring Report shall include a summary of the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual (ODCM) as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979.

In the event that some results are not available prior to May 1 of the year, the report shall be submitted, noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The report shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps[1] covering all sampling locations keyed to a table giving distances and directions from the centerline of the reactor;

[1] One map shall cover stations near the site boundary; a second shall include the more distant stations.

6.9.C.2 Annual Radiological Environmental Monitoring Report (Continued)

discussion of all deviations from the sampling schedule of Table 8.1-1; and discussion of all analyses in which the lower limits of detection (LLD) required by Table 8.1-4 were not achievable.

3. Offsite Dose Calculation Manual (ODCM)

Any changes to the ODCM shall be submitted to the Commission in the semiannual radioactive effluent release report.

OPERATIONAL OBJECTIVES

7.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

7.1 Monitoring Program

Applicability:

At all times.

Specification:

A. ENVIRONMENTAL MONITORING

An environmental monitoring program shall be conducted to evaluate the effects of station operation on the environs and to verify the effectiveness of the source controls on radioactive materials.

The radiological environmental monitoring program shall be conducted as specified in Table 8.1-1.

Action:

1. With the radiological environmental monitoring program not being conducted as specified in Table 8.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Monitoring Report required by Specification 6.9.C.2, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

2. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 7.1-1 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken

SURVEILLANCE REQUIREMENTS

8.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

8.1 Monitoring Program

Specification:

A. ENVIRONMENTAL MONITORING

The radiological environmental monitoring samples shall be collected pursuant to Table 8.1-1 from the specific locations given in the table and figure(s) in the Offsite Dose Calculation Manual (ODCM) and shall be analyzed pursuant to the requirements of Table 8.1-1 and the detection capabilities required by Table 8.1-4.

1. Cumulative dose contributions for the current calendar year from radionuclides detected in environmental samples shall be determined in accordance with the methodology and parameters in the ODCM. These results will be reported in the Annual Radiological Environmental Monitoring Report.

7.1.A ENVIRONMENTAL MONITORING (Continued)

to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 7.2, 7.3, and 7.4. When more than one of the radionuclides in Table 7.1-1 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 7.1-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a member of the public is equal to or greater than the calendar year limits of Specifications 7.2, 7.3, and 7.4. 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 Monitoring Report.

3. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 8.1-1, identify locations for obtaining replacement samples and add them to the Radiological Environmental Monitoring Program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program.

OPERATIONAL OBJECTIVES	SURVEILLANCE REQUIREMENTS
7.1.A ENVIRONMENTAL MONITORING (Continued) Pursuant to Specification 6.9.C.2, identify the cause of the unavailability of samples and identify the new location(s) obtaining replacement samples in the next Annual Environmental Radiation Monitoring Report and also include in the report the table for the ODCM reflecting the new location(s).	
B. LAND USE CENSUS A land use census shall be conducted and shall identify, within a distance of 8 km (5 miles), the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify, within a distance of 5 km (3 miles), the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation. Action 1. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 8.4.A, identify the new location(s) in the next Annual Environmental Radiological Monitoring Report.	B. LAND USE CENSUS The land use census shall be conducted during the growing season, at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Monitoring Report. Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of the two different direction sectors with the highest predicted D/Qs, in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 8.1-1 shall be followed, including analysis of control samples.

OPERATIONAL OBJECTIVES	SURVEILLANCE REQUIREMENTS
7.1.B LAND USE CENSUS (Continued) 2. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 7.1, add the new location(s) to the Radiological Environmental Monitoring Program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Identify the new location(s) in the next Annual Environmental Radiological Monitoring Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).	
7.2 Dose - Liquids Applicability: At all times. Specification: A. The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released at and beyond the site boundary shall be limited: 1. During any calendar quarter to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and 2. During any calendar year to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.	8.2 Dose - Liquids Specification: A. Dose Calculations - Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM for each calendar month during which releases occurred.

OPERATIONAL OBJECTIVES	SURVEILLANCE REQUIREMENTS
7.2 Dose - Liquids (Continued) Action With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a special report that identifies the cause(s), corrective actions taken, and corrective actions to be taken.	
7.3 Dose - Noble Gases Applicability: At all times. Specification: A. The air dose in areas at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited to the following: 1. During any calendar quarter, to $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation; and 2. During any calendar year, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation. Action With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, a special report which identifies the cause(s), the corrective actions taken, and corrective actions to be taken.	8.3 Dose - Noble Gases Specification: A. Dose Calculations - Cumulative dose contributions for the total time period shall be determined in accordance with the ODCM for each calendar month during which releases occurred.

OPERATIONAL OBJECTIVES	SURVEILLANCE REQUIREMENTS
7.4 Dose - Iodine-131, Iodine-133, Radioactive Material in Particulate Form, and Tritium	8.4 Dose - Iodine-131, Iodine-133, Radioactive Material in Particulate Form, and Tritium
Applicability: At all times	Specification: A. Dose Calculations - Cumulative dose contributions for the total time period shall be determined for iodine-131, iodine-133, radioactive material in particulate form with half-lives greater than 8 days, and tritium in accordance with the ODCM for each calendar month during which releases occurred.
Specification: A. The dose to a member of the public from iodine-131, iodine-133, radioactive materials in particulate form with half-lives greater than 8 days, and tritium in gaseous effluents released to areas at and beyond the site boundary shall be limited to the following: 1. During any calendar quarter to $\leq$ 7.5 mrem to any organ, and 2. During any calendar year to $\leq$ 15 mrem to any organ. Action With the calculated dose from the release of iodine-131, iodine-133, radioactive materials in particulate form, and tritium in gaseous effluents exceeding any of the above limits; prepare and submit to the Commission within 30 days, a special report which identifies the cause(s), corrective actions taken, and the corrective actions to be taken.	
7.5 Total Dose	8.5 Total Dose
Applicability: At all times.	Specification: A. Dose Calculations - Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 7.2.A, 7.3.A, and 7.4.A; and in accordance with the ODCM.
Specification: A. The dose or dose commitment to any member of the public from Pilgrim Station sources is limited to $\leq$ 25 mrem to the total body or any organ (except the thyroid, which	

7.5 Total Dose (Continued)

is limited to ≤ 75 mrem) over a period of any calendar year.

Action

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 7.2.A, 7.3.A, or 7.4.A; prepare and submit a special report to the Commission and limit the subsequent releases such that the dose or dose commitment to any member of the public from all uranium fuel cycle sources is limited to ≤ 25 mrem to the total body or any organ (except thyroid, which is limited to ≤ 75 mrem) over any calendar year. This special report shall include an analysis which demonstrates that radiation exposures to all members of the public from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR, Part 190 standard. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR, Part 190 standard.

TABLE 7.1-1
REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analysis	Reporting Levels				
	Water (pCi/L)	Airborne Particulate or Gases (pCi/M^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Vegetables (pCi/kg, wet)
H-3	2×10^4				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-140	2×10^2			3×10^2	

TABLE 8.1-1
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
AIRBORNE			
Particulates	11 Locations (See Table 8.1-2)	Continuous sampling over one week	Gross beta radioactivity 24 hours or more after filter change[1]
Quarterly	11 Locations (See Table 8.1-2)		Composite (by location) for gamma isotopic[2]
Radioiodine	11 Locations (See Table 8.1-2)	Continuous sampling with canister collection weekly	Analyze weekly for I-131
DIRECT[9]	40 Locations (See Table 8.1-3)	Quarterly	Gamma exposure quarterly
	Plymouth Beach and Priscilla/White Horse Beach	Annually	Gamma exposure survey[9]
WATERBORNE (Surface Water)	Discharge Canal	Continuous composite sample	Gamma isotopic[2] monthly, and composite for H-3 analysis quarterly[3]
	Bartlett Pond (SE-1.7 mi)	Weekly grab sample	
	Powder Point (NNW-7.8 mi)[4]	Weekly grab sample	
AQUATIC			
Shellfish (clams, mussels or quahogs as available)	Discharge outfall Duxbury Bay Manomet Point Plymouth or Kingston Harbor Marshfield[4]	Quarterly (at approximate 3-month intervals)	Gamma isotopic[2,6]

TABLE 8.1-1 (Continued)
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
Lobster	Vicinity of discharge point	Four times per season	Gamma isotopic[2] on edible portions
	Offshore[4]	Once per season	
Fish	Vicinity of discharge point	Quarterly (when particular species available) for Groups I and II[5], in season for Groups III and IV[5], annually for each group	Gamma isotopic[2] on edible portions[5]
	Offshore[4]		
Sediments	Rocky Point Plymouth Harbor Duxbury Bay Plymouth Beach Manomet Point Marshfield	Semiannually	Gamma isotopic[2,3,7]
INGESTION (Terrestrial)			
Milk	Plymouth County Farm, when available (W-3.5 mi)[8]	Semimonthly during periods when animals are on pasture, otherwise monthly	Gamma isotopic[2], radio-iodine analysis all samples
	Whitman Farm (NW-21 mi)[4]		
Cranberries	Manomet Point Bog (SE-2.6 mi)	At time of harvest	Gamma isotopic[2] on edible portions
	Bartlett Rd. Bog (SSE/S-2.8 mi)		
	Pine St. Bog (WNW-17 mi)[4]		

TABLE 8.1-1 (Continued)
OPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway or Sample Type	Locations (Direction-Distance) from Reactor	Sampling and Collection Frequency	Type and Frequency of Analysis
Tuberous and green leafy vegetables	Plymouth County Farm (W-3.5 mi)[8] Bridgewater Farm (W-20 mi)[4]	At time of harvest	Gamma isotopic[2] on edible portions
Beef Forage	Plymouth County Farm (W-3.5 mi)[8] Whitman Farm (NW-21 mi)[4]	Annually	Gamma Isotopic[2]

TABLE 8.1-1 (Continued)
NOTES

[1] If gross beta radioactivity is greater than 10 times the control value, gamma isotopic will be performed on the sample.

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

[3] If integrated gamma activity (less K-40) is greater than 10 times the control value (less K-40), strontium-90 analysis will be performed on the sample.

[4] Indicates control location.

[5] Fish analyses will be performed on a minimum of 2 sub-samples, consisting of approximately 400 grams each from each of the following groups:

I. Bottom Oriented	II. Near Bottom Distribution	III. Anadromous	IV. Coastal Migratory
Winter flounder	Tautog	Alewife	Bluefish
Yellowtail founder	Cunner	Rainbow smelt	Atlantic herring
	Atlantic cod	Striped bass	Atlantic menhaden
	Pollock		Atlantic mackerel
	Hakes		

[6] Mussel samples from four locations (immediate vicinity of discharge outfall, Manomet Point, Plymouth or Kingston Harbor, and Green Harbor in Marshfield) will be analyzed quarterly as follows:

One kilogram wet weight of mussel bodies, including fluid within shells will be collected. Bodies will be reduced in volume by drying at about 100°C. Sample will be compacted and analyzed by Ge(Li) gamma spectrometry or alternate technique, if necessary, to achieve a sensitivity of 5 pCi/kg for Cs-134, Cs-137, Co-60, Zn-65, and Zr-95; and 15 pCi/kg for Ce-144. Sensitivity values are to be determined in accordance with a 95% confidence level on k_a and a 50% confidence level on k_b (See HASL-300 for definitions).

The mussel shell sample from one location will be analyzed each quarter. One additional mussel shell sample will be analyzed semiannually. Unscrubbed shells to be analyzed will be dried, processed, and analyzed similarly to the mussel bodies.

TABLE 8.1-1 (Continued)
NOTES

Because of the small volume reduction in pre-processing of shells, sensitivities attained will be less than that for mussel bodies. The equipment and counting times to be employed for analyses of shells will be the same or comparable to that employed for mussel bodies so that the reduction in sensitivities (relative to those for mussel bodies) will be strictly limited to the effects of poorer geometry related to lower sample volume reduction. Shell samples not scheduled for analysis will be reserved (unscrubbed) for possible later analysis.

If radiocesium (Cs-134 and Cs-137) activity exceeds 200 pCi/kg (wet) in mussel bodies, these samples will be analyzed by radiochemical separation, electrodeposition, and alpha spectrometry for radioisotopes of plutonium, with a sensitivity of 0.4 pCi/kg.

[7] Sediment samples from four locations (Manomet Point, Rocky Point, Plymouth Harbor, and head of Duxbury Bay) will be analyzed once per year (preferably early summer) as follows:

Cores will be taken to depths of 30-cm, minimum depth, wherever sediment conditions permit, by a hand-coring sampling device. If sediment conditions do not permit 30-cm deep cores, the deepest cores achievable with a hand-coring device will be taken. In any case, core depths will not be less than 14-cm. Core samples will be sectioned into 2-cm increments; surface and alternate increments will be analyzed, all others will be reserved. Sediment sample volumes (determined by core diameter and/or number of individual cores taken from any single location) and the counting technique will be sufficient to achieve sensitivities of 50 pCi/kg dry sediment for Cs-134, Cs-137, Co-60, Zn-65, and Zr-95 and 150 pCi/kg for Ce-144. In any case, individual core diameters will not be less than 2 inches.

The top 2-cm section from each core will be analyzed for Pu isotopes (Pu-238, Pu-239, and Pu-240) using radiochemical separations, electrodeposition, and alpha spectrometry with target sensitivity of 25 pCi/kg dry sediment. Two additional core slices per year (mid-depth slice from two core samples) will be similarly analyzed.

[8] These locations may be altered in accordance with results of surveys discussed in Specification 8.1.B.

[9] Minimum sensitivities for gamma exposure measurements are as follows:

Gamma exposure - 1 R/hr average exposure rate.
Gamma exposure survey - 1 R/hr exposure rate.

TABLE 8.1-2

AIR PARTICULATES, GASEOUS RADIOIODINE, AND SOIL SURVEILLANCE STATIONS

Sampling Location (Sample Designation)	Distance and Direction from Reactor
Offsite Stations	
East Weymouth (EW) (Control Station)	21 miles NW
Plymouth Center (PC)	4.0 miles W-WNW
Manomet Substation (MS)	2.5 miles SE
Cleft Rock Area (CR)	0.9 miles S
Onsite Stations	
Rocky Hill Road (ER)	0.8 miles SE
Rocky Hill Road (WR)	0.3 miles W-WNW
Overlook Area (OA)	0.03 miles W
Property Line (PL)	0.34 miles NW
Pedestrian Bridge (PB)	0.14 miles N
East Breakwater (EB)	0.35 miles ESE
Warehouse (WS)	0.03 miles SSE

TABLE 8.1-3
EXTERNAL GAMMA EXPOSURE SURVEILLANCE STATIONS[1]

Dosimeter Location (Designation)	Distance and Direction from Station
ONSITE STATIONS	
Property Line (D)	0.17 miles NNW
Property Line (F)	0.12 miles NW
Property Line (I)	0.14 miles W
Property Line (G)	0.20 miles WSW
Rocky Hill Road (A)	0.12 miles SW
Property Line (H)	0.21 miles SSW
Public Parking Area (PA)	0.07 miles N-NNE
Pedestrian Bridge (PB)	0.1 miles NE
Overlook Area (OA)	0.03 miles W
East Breakwater (EB)	0.26 miles ESE
Property Line (C)	3.3 miles ESE-SE
Property Line (HB)	0.34 miles SE
Rocky Hill Road (B)	0.26 miles SSE
Microwave Tower (MT)	0.38 miles S
Emerson Road (EM)	0.68 miles SE-SSE
White Horse Road (WH)	0.89 miles SE-SSE
Property Line (E)	0.75 miles SSE-S
Rocky Hill Road (WR)	0.3 miles W-WNW
Property Line (J)	1.36 miles SSE-S
Property Line (K)	1.42 miles SSE-S
Rocky Hill Road (ER)	0.8 miles SE
Property Line (L)	0.40 miles E

TABLE 8.1-3 (Continued)
EXTERNAL GAMMA EXPOSURE SURVEILLANCE STATIONS[1]

Dosimeter Location (Designation)	Distance and Direction from Station
ONSITE STATIONS (Continued)	
Warehouse (WS)	0.1 miles SE
Property Line (PL)	0.3 miles W
OFFSITE STATIONS	
Duxbury (SS)	6.25 miles SSW-SW
Kingston (KS)	10 miles WNW
North Plymouth (NP)	5.5 miles WNW
Plymouth Center (PC)	4.0 miles W-WNW
South Plymouth (SP)	3 miles WSW
Bayshore Drive (BD)	0.7 miles W-WNW
Cleft Rock Area (CR)	0.9 miles S
Manomet (MP)	2.25 miles ESE-S
Manomet (ME)	2.5 miles SE
Manomet (MS)	2.5 miles SSE
Manomet (MB)	3.5 miles SE-SSE
College Pond (CP)	6.5 miles SSW-SW
Sagamore (CS)	10 miles SSE-S
Plymouth Airport (SA)	8 miles WSW
East Weymouth (EW)[2]	21 miles NW
Saquish Neck (SN)[3]	4.6 miles NNW

[1] Thermal Luminescent Dosimeters (TLDs)
[2] Control Station
[3] TLDs for this location will be provided to a third party and will be analyzed for gamma exposure whenever returned to Boston Edison Company.

TABLE 8.1-4
MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

Analysis	Water (pCi/kg)	Airborne Particulate or Gas (pCi/M^3)	Wet Solids (pCi/kg, wet)	Milk (pCi/g)	Food Products (pCi/kg, wet)	Dry Solids (pCi/kg, dry)
gross beta	4[b]	1×10^{-2}				
^{3}H	2000[d]					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			50
^{65}Zn	30		260			50
^{95}Zr	15					50
^{131}I	1	7×10^{-2}		1	60[c]	
$^{134,137}Cs$	15, 18	1×10^{-2}	130	15	60	50
^{140}Ba	15			15		
^{144}Ce						150

[a] Refer to ODCM for LLD definition.

[b] LLD for surface water.

[c] LLD for leafy vegetables.

[d] If no drinking water pathway exists, a value of 3000 pci/l may be used.

BASES

7/8.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

7/8.1 Monitoring Program

7/8.1.A ENVIRONMENTAL MONITORING

An environmental radiological monitoring program is conducted to verify the adequacy of in-plant controls on the release of radioactive materials. The program is designed to detect radioactivity concentrations to ensure that radiation doses to individuals do not exceed the levels set forth in 10 CFR 50, Appendix I.

A supplemental monitoring program for sediments and mussels has been incorporated into the basic program (see Notes 6 and 7 to Table 8.1-1) as a result of an agreement with the Massachusetts Wildlife Federation. This supplemental program is designed to provide information on radioactivity levels at substantially higher sensitivity levels in selected samples to verify the adequacy (or, alternatively, to provide a basis for later modifications) of the long-term marine sampling schedules. As part of the supplemental program, analysis of mussels for isotopes of plutonium will be performed if radiocesium activity should exceed 200 pCi/kg in the edible portions.

The 200 pCi/kg radiocesium "action level" is based on calculations which show that if radiocesium from plant releases reached this level, plutonium could possibly appear at levels of potential interest.[1] The calculations also show that the dose delivered from these levels of plutonium would not be a significant portion of the total dose attributable to liquid effluents.

The program was also designed to be consistent, wherever applicable, with NUREG 0473.

Groundwater flow at the plant site is into Cape Cod Bay; therefore, terrestrial monitoring of groundwater is not included in this program.

Detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLD). The LLD in Table 8.1.4 is considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), curie, L.A.; "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry", Anal. Chem. 40, 586-93 (1968); and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

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[1] In measurable quantities having a potential dose (human food chain) significance comparable to other nuclides if present at their detection limits.

BASES

7/8.1.B LAND USE CENSUS

This section is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of 10CFR50, Appendix I, Section IV.B.3. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored, since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

7/8.2 DOSE - LIQUID

This section is provided to implement the requirements of Sections II.A, III.A, and IV.A of 10CFR50, Appendix I, to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Because Pilgrim is not a site where plant operations can conceivably affect drinking water, none of these requirements are intended to assure compliance with 40 CFR 141. The dose calculations in the ODCM implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with 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," April 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

BASES

7/8.3 DOSE - NOBLE GASES

This section is provided to implement the requirements of 10CFR50, Appendix I, Sections II.B, III.A, and IV.A to ensure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The surveillance requirements implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the site boundary will be based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

7/8.4 DOSE - IODINE-131, IODINE-133, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND TRITIUM

This section is provided to implement the requirements of Sections II.C, III.A and IV.A of 10 CFR50, Appendix I, to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements of 10CFR50, Appendix I, Section III.A to ensure that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods approved by the NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guides 1.109 and 1.111. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, radioactive material in particulate form with half-lives greater than 8 days, and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in areas at and beyond the site boundary. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

BASES

7/8.5 TOTAL DOSE

This section is provided to meet the dose limitations of 40CFR190 that have now been incorporated into 10CFR20 by 46 FR 18525. The specification requires the preparation and submittal of a special report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of 10CFR50, 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 40CFR190 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 40CFR190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, except dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the limits of 40CFR190, a request for a variance in a special report in accordance with 40CFR190.11 and 10CFR20.405C is considered to be a timely request and fulfills the requirements of 40CFR190 until NRC staff action is completed. This is provided that the release conditions resulting in violation of 40CFR190 have not already been corrected. The variance only relates to the limits of 40CFR190, and does not apply in any way to the other requirements for dose limitation of 10CFR20. An individual is not considered a member of the public during any period in which he/she is engaged in any operation that is part of the nuclear fuel cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Boston Edison Company has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonable achievable. Boston Edison Company filed this information with the Commission by a letter dated May 14, 1985 which requested changes to the Technical Specifications appended to Facility Operating License No. DPR-35 for the Pilgrim Plant. The proposed changes updated those portions of the technical specifications that address radioactive waste management and make them consistent with the current staff positions, as expressed in NUREG-0473. These revised technical specifications will reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63 and 64; and with 10 CFR Part 50, Appendix B.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

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

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant licensees comply with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded. 10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases of radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage, and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the

nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

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

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

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0473 provides radiological effluent technical specifications for boiling water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key

values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment systems.

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

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

(1) Instrumentation

(2) Radioactive effluents

(3) Radiological environmental monitoring

(4) Design features

(5) Administrative controls.

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

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

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

Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

3.0 EVALUATION

The enclosed report (TER-C5506-110) was prepared for us by Franklin Research Center as part of our technical assistance contract program. The report provides their technical evaluation of the compliance of the licensee's submittal with NRC-provided criteria. The staff has reviewed this TER and agrees with the evaluation. Thus, the proposed changes to the radiological effluent technical specifications for Pilgrim Nuclear Power Station have been reviewed, evaluated, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0473 and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes will not remove or relax any existing requirement related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration. Furthermore, the proposed changes will not remove or relax any existing requirement needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Therefore, staff finds the proposed changes acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes in inspection and surveillance requirements. The staff has

determined that the amendment involves no significant increase in the amount and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupation radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need to prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

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

Attachment:

Technical Evaluation Report

Principal Reviewers: WMeinke, CCongel and CWillis

Dated: August 30, 1985

Table 1. Relation Between Provisions of the Regulations and the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors and Boiling Water Reactors

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

Provisions of Title 10 Code of Federal Regulations	Standard Radiological Effluent Technical Specifications: Instrumentation: Rad. Liquid Effl. Monitoring	Instrumentation: Rad. Gas. Effl. Monitoring	Radioactive Effluents: Liquid: Effluent Concentration	Liquid: Dose	Liquid: Radwaste Treatment	Liquid: Holdup Tanks	Gaseous: PWR/BWR: Dose Rate	Gaseous: PWR/BWR: Dose Noble Gases	Gaseous: PWR/BWR: Dose I-131, Trit. and Part.	Gaseous: PWR/BWR: Explosive Gas Mixture	Gaseous: PWR: Gaseous Radwaste Treatment	Gaseous: PWR: Gas Storage Tanks	Gaseous: BWR: Gaseous Radwaste Treatment	Gaseous: BWR: Ventilation Exhaust Treatment	Gaseous: BWR: Main Condenser	Gaseous: BWR: Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Envir. Monitoring: Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Design Features: Site Boundaries*	Administrative Control: Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
§ 50.36a Technical specifications on effluents from nuclear power reactors																												
Remain within limits of § 20.106			●		●		●				●					●												
Establish and follow procedures to control effluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				
Maintain and use radioactive waste system equipment				●	●			●	●		●		●	●													●	
Submit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●
§§ 20.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●						●	
Part 50 Appendix A - General Design Criteria																												
Criterion 60 - Control of releases of radioactive materials to the environment	●	●		●	●					●	●	●	●	●	●	●	●							●			●	
Criterion 61 - Fuel storage and handling and radioactivity control	●	●																										
Criterion 63 - Monitoring fuel and waste storage	●	●																										
Criterion 64 - Monitoring radioactivity releases	●	●													●				●								●	
Part 50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●	
Part 50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																												
Maintain releases within design objectives				●	●			●	●		●		●														●	
Establish surveillance & monitoring program to provide data on:																												
(1) quantities of rad. matls. in effluents	●	●																	●									
(2) radiation & rad. matls. in the environment																				●								
(3) changes in use of unrestricted areas																									●	●		
Exert best efforts to keep releases "ALARA"				●	●			●	●		●		●															●
Submit report if calculated doses exceed the design objective				●				●	●		●		●												●			
Demonstrate conform. to des. obj. by calc. proced.				●				●	●		●		●														●	
Part 100																●												

*Note: Needed to fully implement other specifications.

TECHNICAL EVALUATION REPORT

NRC DOCKET NO. 50-293

NRC TAC NO. 8123

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 4

FRC TASK 110

RADIOLOGICAL EFFLUENT TECHNICAL
SPECIFICATION IMPLEMENTATION (A-2)

BOSTON EDISON COMPANY
PILGRIM NUCLEAR POWER STATION UNIT 1

TER-C5506-110

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

FRC Group Leader: S. Pandey
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June 19, 1985

Prepared by: L. Chen
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Approved by: A P Carfagno
Department Director
Date: 6-19-85

FRANKLIN RESEARCH CENTER
DIVISION OF ARVIN/CALSPAN
20th & RACE STREETS, PHILADELPHIA, PA 19103

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FOREWORD

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

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of Pilgrim Nuclear Station Unit 1 with regard to Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

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

1.2 GENERIC BACKGROUND

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

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

These other issues are specifically stipulated by the following regulations:

- 10CFR20 [4], "Standards for Protection Against Radiation," Paragraphs 20.105(c), 20.106(g), and 20.405(c) require that nuclear power plants and other licensees comply with 40CFR190 [5], "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40CFR190 limits have been or may be exceeded.

- 10CFR50, Appendix A [6], "General Design Criteria for Nuclear Power Plants," contains Criterion 60 - Control of releases of radioactive materials to the environment; Criterion 63 - Monitoring fuel and waste storage; and Criterion 64 - Monitoring radioactivity releases.

- 10CFR50, Appendix B [7], establishes the quality assurance required for nuclear power plants.

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

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

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

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

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

1.3 PLANT-SPECIFIC BACKGROUND

In response to the NRC's request, the Licensee, Boston Edison Company, submitted a RETS proposal dated February 21, 1979 [14] on behalf of Pilgrim Nuclear Power Station Unit 1, which was followed by a submittal of the ODCM [15]. In the RETS submittal, the Licensee had used a non-standard format. In an initial evaluation by the Franklin Research Center (FRC), an independent review team, the Licensee's RETS submittal was evaluated against the model RETS (NUREG-0473, Draft Revision 2 [1]) and assessed for compliance with the stipulated provisions. Review of the ODCM was conducted in accordance with NRC-issued guidelines (NUREG-0133). Copies of the draft review, dated May 7, 1982 [16, 17], were delivered to the NRC and the Licensee prior to a site visit by the reviewers.

The site visit was conducted on June 7-8, 1982 by the reviewers with the participation of plant personnel and the NRC staff. Discussions focused on the initial review of the proposed changes to the RETS and on the technical approaches for an ODCM. The deficiencies in the Licensee's proposed RETS were considered, deviations from NRC guidelines were pointed out, many differences were clarified, and only a few items remained unresolved pending justification by the Licensee. These issues are summarized in Reference 18.

In a letter transmittal dated April 15, 1983 [19], the Licensee submitted to NRC a revised RETS for review. A revised ODCM followed later on June 16,

1983 [20]. Both submittals were reviewed by FRC, and discrepancies were documented [21] and transmitted to NRC.

The final version of the Pilgrim Nuclear Power Station Unit 1 RETS dated May 14, 1985 [22] was submitted to the NRC and transmitted to the FRC reviewers together with explanations provided by the Licensee. These documents were subsequently reviewed. The final evaluation of the RETS was detailed in a comparison report [23] which used NUREG-0473, Draft Revision 3 [13] to evaluate the Licensee's submittal. The comparison report also incorporates NRC comments [24] which serve as additional guidelines regarding plant-specific issues.

Evaluation of Licensee's ODCM is based on the submittal of Reference 20.

2. REVIEW CRITERIA

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

NUREG-0472, RETS for PWRs

NUREG-0473, RETS for BWRs

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

Twelve essential criteria are given for the RETS and ODCM:

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

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

3. Offsite radiation doses of radioactivity shall be ALARA.

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

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

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

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

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

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

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

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

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

Subsequent to the publication of NUREG-0472 and NUREG-0473, the NRC staff issued guidelines [25, 26], clarifications [27, 28], and branch positions [29, 30, 31, 32] establishing a policy that guides the licensees of operating reactors to meet the intent, if not the letter, of the model RETS provisions. The NRC branch positions issued since the RETS implementation review began have clarified the model RETS implementation for operating reactors.

Review of the ODCM was based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual" [33]; NUREG-0133 [9]; and Regulatory Guide 1.109 [34]. The ODCM format is left to the licensee and may be simplified by tables and grid printouts.

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF RADIOLOGICAL EFFLUENT SYSTEM

This section briefly describes the liquid and gaseous effluent radwaste treatment systems, release paths, and control systems installed at Pilgrim Nuclear Power Station Unit 1, a BWR. The plant is located on the western shore of Cape Cod Bay in the town of Plymouth, Plymouth County, Massachusetts.

3.1.1 Radioactive Liquid Effluent

The liquid radioactive wastes at Pilgrim Nuclear Power Station consist of three types: clean wastes, chemical wastes, and miscellaneous wastes. The bulk of clean wastes comes from the reactor coolant, condensate, and feedwater systems. Following processing and batch sampling, the liquid is pumped to the condensate storage tank for reuse in the plant or discharge offsite. The chemical and miscellaneous wastes are checked for activity level following storage and decay and are either released to the circulation water for discharge or returned for further processing. All these wastes are discharged in batches into the discharge canal. Also joined into the circulating water and released to the environment is the continuous discharge of service water and neutralizing sump discharge. The above effluent treatment systems and flow paths are shown in Figure 1.

3.1.2 Radioactive Gaseous Effluent

The offgases from Pilgrim Nuclear Power Station are processed and routed to the plant stack for dilution and discharged to the atmosphere (see Figure 2). The substreams also routed to the plant stack are the gland seal/mechanical vacuum pumps' discharges, and discharges from the turbine building. The gaseous radwaste system consists of the augmented offgas (AOG) treatment system and a delay line for condenser offgas which provides approximately 30 minutes of decay time prior to release via the stack. A separate effluent line from the ventilation system is released via the reactor building vent.

3.2 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

The evaluation of the Licensee's proposed RETS against the provisions of NUREG-0473 included the following: (1) a review of information provided in the Licensee's 1979 and 1982 draft submittals [14, 15], (2) a discussion of problem areas in those submittals by means of a site visit [18], (3) a review of the Licensee's 1983 draft RETS and ODCM submittals [19, 20], and (4) a review of the Licensee's 1985 final RETS submittal [22].

3.2.1 Effluent Instrumentation

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

3.2.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

A radiation monitor has been installed for the liquid radwaste effluent line (Figure 1) which combines effluent streams from the releases of high purity wastes, low purity wastes, and detergent wastes. This monitor at the radwaste effluent line is equipped with an automatic termination function. The Licensee did not provide monitors for batch releases from the neutralizer sump and salt service water, both also merged into the discharge canal. However, the Licensee has made a commitment to perform the sampling and analysis for these two effluent lines before discharges are made. This satisfies the intent of NUREG-0473.

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

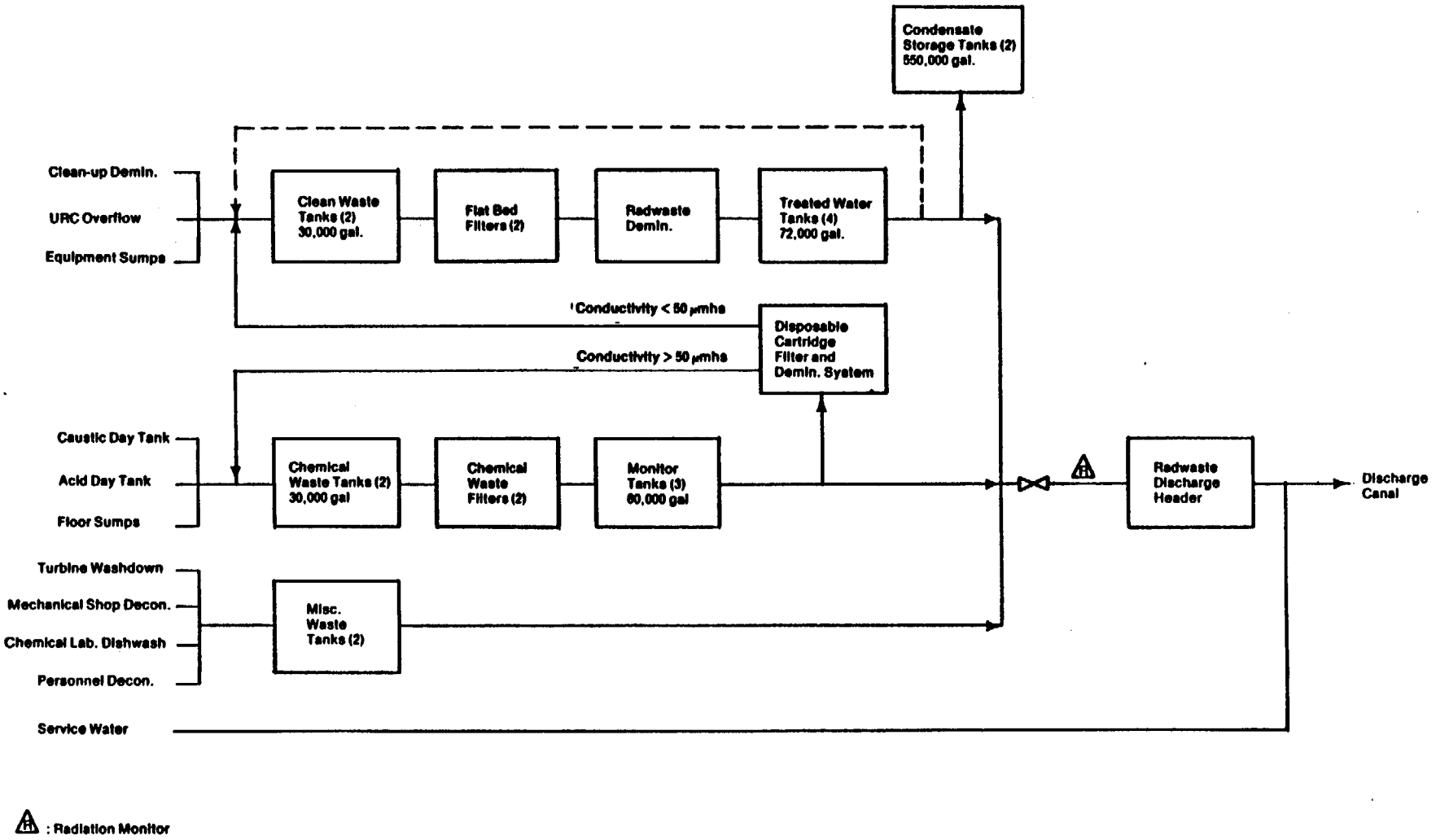

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for Pilgrim Nuclear Power Station Unit 1

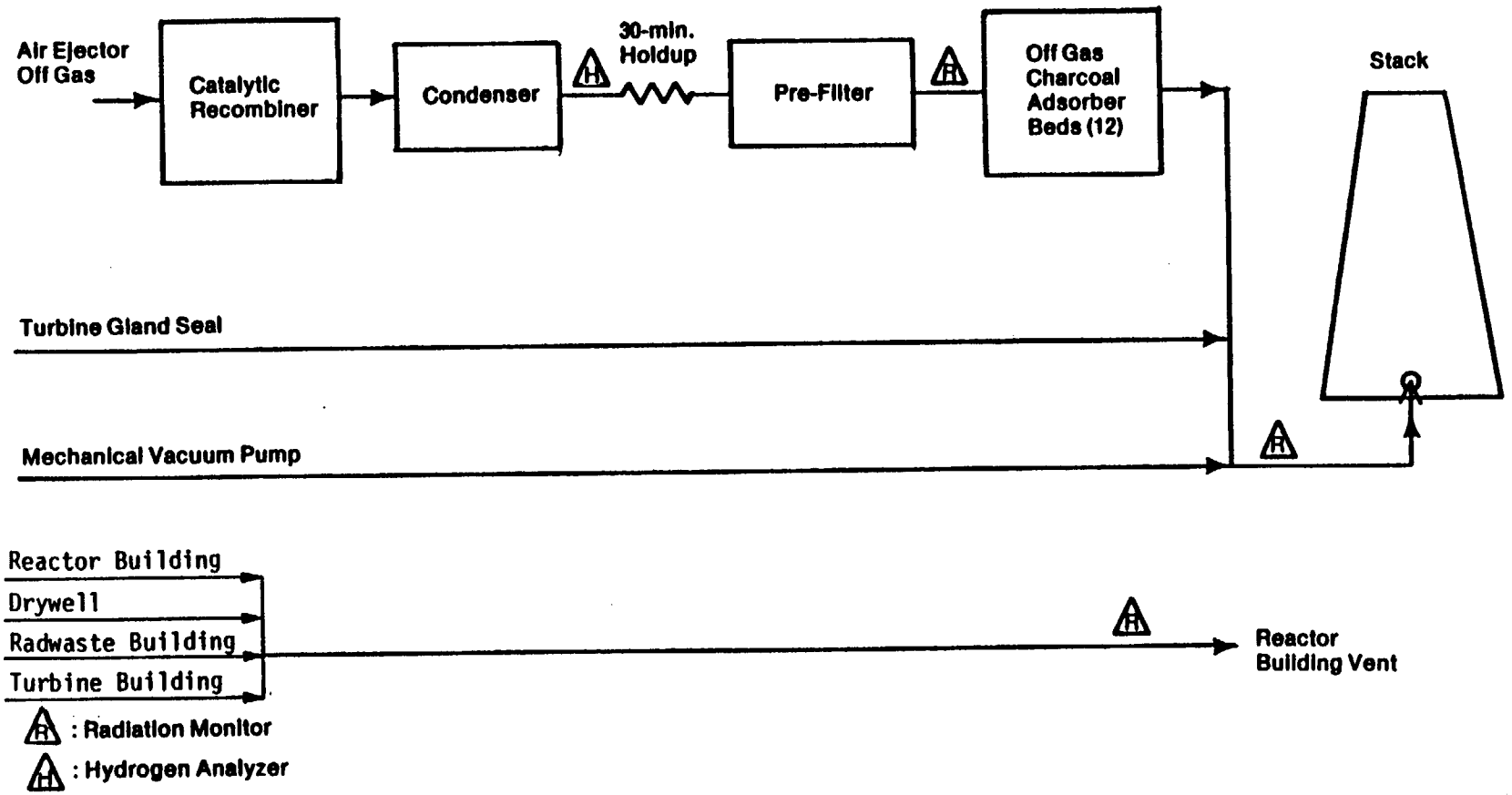

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for Pilgrim Nuclear Power Station Unit 1

3.2.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation

The Licensee has provided noble gas monitors for the air ejector offgas system, the stack gas effluent system, and the reactor building ventilation system, respectively. The monitor at the air ejector has alarm and automatic control functions. Both the stack gas effluent system and the reactor building ventilation system are also equipped with iodine and particulate samplers.

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

3.2.2 Concentration and Dose Rates of Effluents

3.2.2.1 Liquid Effluent Concentration

In Section 3.8 of the Licensee's submittal, a commitment is made to maintain the concentration of radioactive liquid effluents released to areas at or beyond the site boundary to within 10CFR20 limits, and, if the concentration of liquid effluents exceeds these limits, the concentration will be restored without delay to a value equal to or less than the MPC specified in 10CFR20. All batches of radioactive liquid effluents from the release tanks are sampled and analyzed in accordance with a sampling and analysis program (Table 4.8-1 of the Licensee's submittal) which meets the intent of NUREG-0473.

It was determined that the Licensee-proposed specification meets the intent of NUREG-0473 for liquid effluent concentration.

3.2.2.2 Gaseous Effluent Dose Rate

In Section 3.8.D of the Licensee's submittal, a commitment is made to maintain the offsite dose rate from radioactive gaseous effluents to areas at and beyond the site boundary within 10CFR20 limits, or the equivalent dose rate values prescribed by Section 3.11.2.1 of NUREG-0473. If the dose rate of gaseous effluents exceeds these limits, it will be restored without delay to a value equal to or less than these limits. This commitment satisfies the provisions of NUREG-0473.

The radioactive gaseous waste sampling and analysis program (Table 4.8-3 of the Licensee's submittal) provides adequate sampling and analysis of the plant stack discharges, including the substreams, and therefore meets the intent of NUREG-0473.

3.2.3 Offsite Doses from Effluents

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

3.2.4 Effluent Treatment

The objectives of the RETS with regard to effluent treatment are to ensure that wastes are treated to keep releases ALARA and to satisfy the provisions of technical specifications governing the maintenance and use of radwaste treatment equipment.

In the proposed Specifications 3.8.C and 3.8.F, the Licensee has made a commitment to use the liquid and gaseous radwaste treatment systems to reduce the radioactive materials in liquid and gaseous waste prior to their discharge. The commitment includes:

a. Liquid radwaste treatment system. The Licensee proposes to treat the liquid effluents whenever the averaged monthly dose exceeds the limits of 0.06 mrem to the whole body or 0.2 mrem to any organ.

b. Gaseous radwaste treatment system. The Licensee proposes to treat the gaseous effluents by operating the augmented offgas system to meet the proposed Specification 3.8.F.

It was determined [24] that the Licensee's proposal on the radwaste treatment system meets the intent of NUREG-0473.

3.2.5 Radioactivity Inventory Limits

The objective of the RETS with regard to the liquid tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10CFR20 for nonoccupational exposure. The Licensee did not include such a liquid tank specification in the proposed submittal, since there are no unprotected outdoor radwaste tanks on the Pilgrim site. For radioactive releases at the steam jet air ejector, a release rate limit of 500,000 microcuries/sec has been set for noble gases. The Licensee has made a commitment to restore the gross radioactivity rate to within its limit within 72 hours or to place the plant in hot standby within the next 12 hours, if the release rate exceeds the proposed limits. The Licensee's proposals on the radioactivity inventory limits meet the intent of NUREG-0473.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in waste gas systems. The Licensee stated that the waste gas holdup system is designed to withstand a hydrogen/oxygen explosion.

In Specification 3.8.F of the Licensee's submittal, a commitment is made to limit the concentration of hydrogen in the augmented offgas treatment system to less than 2% by volume. The Licensee has provided one hydrogen monitor to continuously monitor for the possibility of an explosive gas mixture in the waste gas holdup system. The Licensee action specification states that the hydrogen concentration will be restored within 48 hours if concentration levels are found to exceed the trigger levels.

As in the previous reviews of other plants, it determined that the Licensee meets the intent of NUREG-0473 in the interim until NRC completes its study of the explosive gas problem.

3.2.7 Solid Radwaste System

The objective of the RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before it is shipped to the burial site. Specification 3.11.3 of NUREG-0473 provides for the establishment of a PCP to show compliance with this objective. The Licensee stated that the methods used to accomplish the processing and packaging of solid radioactive waste material are procedurally addressed. These procedures are required by the Licensee's existing Technical Specification Section 6.8, and are reviewed and approved by the appropriate committees per Technical Specification 6.5.A.6. This was determined to meet the intent of NUREG-0473 [24].

3.2.8 Radiological Environmental Monitoring Program

The objectives of the RETS with regard to environmental monitoring are to ensure that an adequate and full-area-coverage monitoring program exists and that the 10CFR50, Appendix I requirements for technical specifications on environmental monitoring are satisfied. In all cases, the Licensee has followed NUREG-0473 guidelines, including the Branch Technical Position dated November 1979 [30], and has provided an adequate number (40) of thermo-luminescent dosimeter (TLD) sample locations omitting water sectors not applicable to land pathways. The Licensee's methods of analysis and maintenance of yearly records satisfy the NRC guidelines and meet the intent of 10CFR50, Appendix I. The Licensee has also made a commitment to document the environmental monitoring sample locations in the ODCM, which meets the intent of NUREG-0473. The specification for the land use census satisfies the provisions of Section 3.12.2 of NUREG-0473 by providing for an annual census in the specified areas. Analysis of site environmental media is conducted for

the Licensee by the Yankee Atomic Environmental Laboratory (YAEL), which participates in an NRC-approved interlaboratory comparison program, thus meeting the intent of NUREG-0473 [24].

It is therefore concluded that the radiological environmental monitoring program as proposed by the Licensee meets the intent of NUREG-0473.

3.2.9 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The Licensee's administrative structure designates the Nuclear Safety Review and Audit Committee (NSRAC) for review and audit of the changes to the ODCM. The Licensee stated that audits of the radiological monitoring program and the quality assurance program under Regulatory Guide 4.1 are encompassed under existing Specification 6.5.B.8.5. The Licensee further stated that the ODCM will be included as a reference in the audit scope of the radiological monitoring program. The processing and packaging of the solid radwate materials are covered by the Licensee's existing Technical Specification Section 6.8, which is subject to review and audit by the appropriate committees. It was determined [24] that the Licensee-proposed administrative structure satisfactorily meets the intent of NUREG-0473.

3.2.10 Procedures and Records

The objective of the RETS with regard to procedures is to satisfy the provisions for written procedures specified in NUREG-0473. It is also an objective of RETS to properly retain the documented records related to the environmental monitoring program and certain QA procedures.

The Licensee's existing procedures and the newly proposed RETS specifications were deemed [24] to meet the intent of NUREG-0473.

3.2.11 Reports

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

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

1. Annual radiological environmental operating report. In Section 6.9.C.2 of the Licensee's submittal, a commitment was made to provide an annual radiological environmental operating report that includes summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities. The report also includes the results of land use censuses, specified by Specification 7.1.B of the submittal.

2. Semiannual radioactive release reports. In Section 6.9.C.1 of the Licensee's submittal, a commitment was made to provide semiannual effluent release reports which include the quantity of each of the principal radionuclides released at and beyond the site boundary in liquid and gaseous effluents during the reporting period. This commitment was considered [18] as meeting the intent of NUREG-0473.

3. Special report. The Licensee has made a commitment to file a 30-day special report to the NRC under the following conditions as prescribed by the proposed specifications:

   - o Exceeding radioactive liquid effluent limits according to:

     Dose, Specifications 7.2.A and 3.8.C

   - o Exceeding radioactive gaseous effluent limits according to:

     Dose, Specifications 7.3.A, 7.4.A, and 3.8.F

   - o Exceeding radioactive effluent limits according to:

     Total Dose Commitment, Specification 7.5

   - o Exceeding the reporting levels of Table 7.1-1 for the radioactivity measured in the environmental sampling medium, Specification 7.1.

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

3.2.12 Implementation of Major Programs

One objective of the administrative controls is to ensure that implementation of major programs such as the ODCM, PCP, and major changes to the radioactive waste treatment system follow appropriate administrative procedures. The Licensee has made a commitment to review, report, and implement the ODCM. The Licensee considers that implementation of programs, such as processing of solid radwaste, and major changes to the radwaste treatment systems are covered under the existing Technical Specifications. These commitments meet the intent of NUREG-0473 [24].

3.3 OFFSITE DOSE CALCULATION MANUAL

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

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

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

3.3.1 Evaluation

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

The Licensee demonstrated the method of calculating the radioactive liquid concentration by describing in the ODCM the means of collecting and analyzing representative samples prior to and after releasing liquid effluents into the circulating water discharge.

Methods are also included for showing that dose rates released to unrestricted areas due to noble gases, radioiodines, and particulates with half-lives greater than 8 days are in compliance with 10CFR20. In this calculation, the Licensee has considered effluent releases from the plant stack. The Licensee has demonstrated that the described methods and relevant parameters have generally followed the conservative approaches provided by NUREG-0133 and Regulatory Guide 1.109.

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

For liquid releases, the Licensee has identified aquatic foods (fish and shellfish) ingestion and shoreline deposits as the viable pathways. In the calculation, the Licensee has followed the suggested methodology given in Regulatory Guide 1.109. The Licensee has used the maximally exposed individual as the reference receptor. To correctly assess the cumulative dose, the Licensee intends to estimate the dose once per 31 days.

Evaluation of the cumulative dose from noble gas releases includes both beta and gamma and air doses at and beyond the site boundary. The critical organs under consideration are the total body and skin for gamma and beta radiation, respectively. Again, the Licensee has generally followed the methodology of NUREG-0133 and Regulatory Guide 1.109.

For radioiodines and particulates with half-lives greater than 8 days, the Licensee has provided a method to demonstrate that cumulative doses calculated from the release meet both quarterly and annual design objectives.

The Licensee has provided flow diagrams (Figures 8-1 and 8-2 of the proposed submittal) for both the liquid and gaseous waste systems and has identified the effluent paths and components of the radioactive liquid and gaseous waste treatment systems.

The Licensee has provided a description of sampling locations (Tables 7-2 and 7-3 of the proposed submittal) in the ODCM. The Licensee also included maps and detailed locations in the submittal.

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

4. CONCLUSIONS

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

1. The Licensee's proposed Radiological Effluent Technical Specifications (RETS) submitted May 14, 1985 [22] meet the intent of NUREG-0473, "Radiological Effluent Technical Specifications."

2. The Licensee's Offsite Dose Calculation Manual (ODCM) submitted June 16, 1983 [20] uses documented and approved methods that are generally consistent with the criteria of NUREG-0133 and applicable to Pilgrim Nuclear Power Station Unit 1. It is thus an acceptable reference.

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

	Technical Specifications		Replaces or Updates Existing Tech. Specs. (Section)	Evaluation
	NRC Staff Std. RETS NUREG-0473 (Section)*	Licensee Proposal (Section)		
Effluent Instrumentation	3/4.3.3.3.10 3/4.3.3.3.11	3.8.B 3.8.E	6.4.1 Appendix A	Meets the intent of NRC criteria
Radioactive Effluent Concentrations	3/4.11.1.1 3/4.11.2.1	3.8.A 3.8.D	3.8.A 3.8.B	Meets the intent of NRC criteria
Offsite Doses	3/4.11.1.2, 3/4.11.2.2, 3/4.11.2.3, 3/4.11.4	7.2 7.3 7.4 7.5	To be added to Appendix A	Meets the intent of NRC criteria
Effluent Treatment	3/4.11.1.3 3/4.11.2.4	3.8.C 3.8.F	3.8.A 3.8.B	Meets the intent of NRC criteria
Radioactivity Inventory Limits	3/4.11.1.4 3/4.11.2.6	NA 3.8.G	Not addressed 3.8.B	Meets the intent of NRC criteria
Explosive Gas Mixtures	3/4.11.2.5B	3.8.F	To be added to Appendix A	Meets the intent of NRC criteria in the interim
Solid Radioactive Waste	3/4.11.3	NA	Not addressed	Meets the intent of NRC criteria
Environmental Monitoring	3/4.12.1	7.1	4.8.D	Meets the intent of NRC criteria
Audits and Reviews	6.5.1, 6.5.2	6.5.1, 6.5.2	6.5.1, 6.5.2	Meets the intent of NRC criteria
Procedures and Records	6.8, 6.10	6.8, 6.10	6.8, 6.10	Meets the intent of NRC criteria
Reports	6.9	6.9	6.9	Meets the intent of NRC criteria
Implementation of Major Programs	6.13, 6.14, 6.15	6.9.C.3	To be added to Appendix A	Meets the intent of NRC criteria

*Section number sequence is according to NUREG-0473, Rev. 3, Draft 7" [13].

5. REFERENCES

1. "Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0473

2. Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"

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

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

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

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

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

8. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0472

9. "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, A Guidance Manual for Users of Standard Technical Specifications"
   NRC, October 1978
   NUREG-0133

10. C. Willis and F. Congel (NRC)
    "Summary of Draft Contractor Guidance of RETS"
    Presented at the AIF Environmental Subcommittee Meeting, Washington, DC
    May 19, 1982

11. F. Congel (NRC)
    Memo to RAB Staff (NRC)
    Subject: Interim Changes in the Model Radiological Effluent Technical Specifications (RETS)
    August 9, 1982

12. W. Meinke (NRC)
Memo to M. Strum (Yankee Atomic Electric Company)
Subject: BWR-Specific Changes for NUREG-0472/3, Rev. 3
September 20, 1982

13. "Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 3, Draft 7", intended for contractor guidance in reviewing RETS proposals for operating reactors
NRC, December 1983
NUREG-0473

14. G. C. Andognini (Boston Edison Company)
Letter of Transmittal to NRC
Subject: Pilgrim Nuclear Power Station - Proposed Technical Specifications Changes Related to Radiological Effluents
February 21, 1979
NRC Docket No. 50-293

15. G. C. Andognini (Boston Edison Company)
Letter of Transmittal to NRC
Subject: Pilgrim Nuclear Power Station - Proposed Offsite Dose Calculation Manual
March 13, 1979
NRC Docket No. 50-293

16. "Comparison of Specification NUREG-0473, Radiological Effluent Technical Specifications for BWRs, vs. Licensee Submittal of Radiological Effluent Technical Specifications for Pilgrim Nuclear Power Station Unit 1" (Draft)
Franklin Research Center, May 7, 1982

17. Technical Review of Offsite Dose Calculation Manual for Pilgrim Nuclear Station Unit 1 (Draft)
Franklin Research Center, May 7, 1982

18. Franklin Research Center
Letter of Transmittal to NRC
Subject: Trip report on site visit to Pilgrim Nuclear Power Station Unit 1
June 9, 1982

19. W. D. Harrington (Boston Edison Company)
Letter of Transmittal to NRC
Subject: Pilgrim Plant - Radiological Effluent and Environmental Monitoring Technical Specifications (RETS)
April 15, 1983
NRC Docket No. 50-293

20. W. D. Harrington (Boston Edison Company)
Letter of Transmittal to NRC
Subject: Pilgrim and Palisades Plants - Offsite Dose Calculation Manual (ODCM)
June 16, 1983
NRC Docket No. 50-293

21. S. Pandey/S. Chen
Informal Technical Communication to NRC
Subject: Attachment 1, Comments on Pilgrim's ODCM Submittal (dated June 16, 1983); Attachment 2, Comparison Sheets on Pilgrim RETS Submittal (dated April 15, 1983)
April 9, 1984

22. W. D. Harrington (Boston Edison Company)
Letter of Transmittal to NRC
Subject: Pilgrim Plant - Radiological Effluent Technical Specifications (RETS)
May 14, 1985
NRC Docket No. 50-293

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