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AUG 1 1972

Docket No. 50-271

Change No. 2
License No. DPR-28

Vermont Yankee Nuclear Power Corp.
ATTN: Mr. Albert A. Cree
President
77 Grove Street
Rutland, Vermont 05701

Gentlemen:

The enclosed changes to the Technical Specifications of Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station are being made to clarify a number of requirements, including the clarification requested in Mr. Vandenburgh's letter dated June 29, 1972, and to correct inconsistencies, omissions, and errors, which have been revealed since the Technical Specifications were originally issued. These changes do not present significant safety considerations not described or implicit in the Safety Analysis Report and there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station are changed by removing the pages numbered 26, 50, 52, 122, 123, 130, 131, 132, 133, 141, 146, 184, 185, 186, 187, 188, and 189 (all with dates prior to 7/28/72), and replacing them with the enclosed revised pages numbered 26, 50, 52, 122, 123, 130, 131, 132, 132a, 133, 141, 146, 184, 185, 186, 187, 188, 189, 189a that are dated 7/28/72.

Sincerely,

Original Signed by
Roger S. Boyd

Roger S. Boyd, Assistant Director
for Boiling Water Reactors
Directorate of Licensing

Enclosure:
Revised pages to Technical
Specifications

cc: See attached

cc: John A. Ritsher, Esquire
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OFFICE ▶	L:BWR-1	L:BWR-1	L:BWR-1	L:AD:BWR		
SURNAME ▶	WMinners:lrk	[illegible]sley	WRButler	RSBoyd		
DATE ▶	7/31/72	7/ /72	7/31/72	7/31/72		

Notes: Table 4.1.1

1. Definitions for Instrument Check, Instrument Functional Test, and Instrument Calibration are given in Section 1 of this specification. Functional tests and calibrations are not required when the channels are not required to be operative or are tripped. If tests or calibrations are missed, they shall be performed prior to returning the instrument channels to an operable status.

2. A description of the three groups is included in the bases of this specification.

3. Initially once per month; thereafter, with a longer interval determined by the test results on this type of instrumentation.

4. An instrument check will be performed on low reactor water level once per day and on high steam line radiation once per shift.

5. The water level in the reactor vessel will be perturbed and the corresponding level indicator changes will be monitored. This perturbation test will be performed every month after the completion of the monthly functional test program.

6. Physical inspection of these position switches will be performed during the refueling outages.

7. An instrument alignment shall be performed every 3 months using a standard current source. Calibration using a radiation source shall be made during each refueling outage.

8. The frequency of LPRM calibrations shall be at least every 1000 effective full power hours; full power is defined as 1593 MWt.

9. The 90 percent setdown trip in $\leq$ 30 seconds will be tested and calibrated in conjunction with the high flux test.

10. Calibrate and test prior to every start-up, if not done in the previous week. Check once per shift and calibrate weekly when the plant is in the mode in which this function must be operating.

11. To verify independence of the safety system circuitry, this functional test will be performed at every refueling and, during operation, following maintenance of the associated circuitry.

12. Calibrate and test within one week after each refueling startup and at six month intervals, thereafter.

TABLE 4.2.1

MINIMUM TEST & CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Core Spray System			
Trip Function	Functional Test	Calibration	Instrument Check
High Drywell Pressure	(Note 1)	every 3 months	---
Low-Low Reactor Vessel Water Level	(Note 1)	every 3 months	---
Low Reactor Pressure	(Note 1)	every 3 months	---
Pump 14-1A, Discharge Press	(Note 1)	every 3 months	---
Auxiliary Power Monitor	(Note 1)	every refueling	---
Pump Bus Power Monitor	(Note 1)	none	--
Sparger High Pressure	(Note 1)	every 3 months	---
Trip System Logic except relays 14A-K11A 14A-K11B 14A-K19A 14A-K19B	every 6 months (Note 2)	every 6 months (Note 3)	---

TABLE 4.2.1 (CONT)

Low Pressure Coolant Injection System (Cont)			
Trip Function	Functional Test	Calibration	Instrument Check
Auxiliary Power Monitor	(Note 1)	every refueling	---
Pump Bus Power Monitor	(Note 1)	none	---
Trip System Logic except relays 10A-K29A, 10A-K29B, 10A-K30A, 10A-K30B, 10A-K38A, 10A-K38B, 10A-K42A, 10A-K42B	every 6 months (Note 2)	every 6 months (Note 3)	---

High Pressure Coolant Injection System			
Trip Function	Functional Test	Calibration	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	every 3 months	---
Condensate Storage Tank Low Water Level	(Note 1)	every 3 months	---
High Drywell Pressure	(Note 1)	every 3 months	---
Suppression Chamber High Water Level	(Note 1)	every 3 months	---
Bus Power Monitor	(Note 1	none	once each shift
Trip System Logic	every 6 months (Note 2)	every 6 months (Note 3)	---

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

d. Type A Retest Schedule

(1) After the initial preoperational leakage rate test, two Type A tests shall be performed at approximate equal intervals between the inservice inspection shutdowns conducted at 10-year intervals throughout the service lifetime of the plant. In addition, a Type A test shall be performed at the end of the 10-year interval and may coincide with the inservice inspection shutdown.

e. Type B Tests (Penetrations with gasketed seals, expansion bellows, or other type or resilient seals)

(1) Penetrations and seals listed in Table 4.7.1 shall be leak tested at 44 psig once during every reactor operating cycle.

(2) Airlocks shall be tested at 44 psig at 4-month intervals except when air locks are not opened during this interval, in which case tests shall be performed after each opening, but no interval shall be longer than one operating cycle.

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

f. Acceptance Criteria and Corrective Action for Type B Tests

(1) The sum of the leakage from all penetrations (Types B & C) shall not exceed 60% of L_p. Repair and retest will be conducted to insure compliance, if necessary.

g. Type C tests (operability and leak tightness tests on those isolation valves on lines which penetrate the containment boundary and perform a containment function).

(1) At least once per reactor operating cycle the isolation valves listed in Table 3.7.1a shall be tested for leakage rate measurement.

(2) Prior to violating the integrity of a system outside the primary containment, which is connected to any valve listed in Table 3.7.1.b, the isolation valves bounding the opening shall be tested for leakage rate. If the opening cannot be isolated from the containment by two isolation valves which meet the acceptance criteria of 4.7.A.2.h, a blank flange shall be installed on the opening.

h. Acceptance Criteria and Corrective Action for Type C Tests

(1) The sum of the leakage from all penetrations (Types B & C) shall

TABLE 3.7.1.a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A,D & 2-86A,D)	4	4	5(note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9, 16-19-8)		2	10	Closed	SC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Closed	SC
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Closed	SC
3	Containment Purge Supply (16-19-23)		1	10	Closed	SC
3	Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22B)		3	NA	Closed	SC
5	Reactor Cleanup System (12-15, 12-18)	1	1	25	Open	GC
5	Reactor Cleanup System (12-68)		1	45	Open	GC
6	HPCI (23-15, 23-16)	1	1	55	Open	GC
6	RCIC (13-15, 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process

Table 3.7.1.b

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES NOT SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A, B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A, B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A, B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A, B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A, B)		2	20	Open	GC
3	Containment Air Sampling System (109-75A, D; 1, 2 109-176A, B)		10	5	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
4	RHR Reactor Head Cooling (10-32, 10-33)	1	1	25	Closed	SC
	Feedwater Check Valves (2-28 A, B)	2	2	NA	Open	Process
	Control Rod Hydraulic Return Check Valves (3-110, 3-113)	1	1	NA	Open	Process
	Reactor Head Cooling Check Valve (10-29)	1		NA	Closed	Process
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Process

Table 3.7.1 Notes:

1. Isolation signals are as follows:

Group 1: The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor low-low water level
2. Main steam line high radiation
3. Main steam line high flow
4. Main steam line tunnel high temperature
5. Low main steam line pressure (run mode only)

Group 2: The valves in Group 2 are closed upon any one of the following conditions:

1. Reactor low water level
2. High drywell pressure

Group 3: The valves in Group 3 are closed upon any one of the following conditions:

1. Reactor low water level
2. High drywell pressure
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor

Group 4: The valves in Group 4 are closed upon any one of the following conditions:

1. Reactor low water level
2. High drywell pressure
3. High reactor pressure

Group 5: The valves in Group 5 are closed upon low reactor water level.

Group 6: The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line. The signals indicating a steam line break for the respective steam line are as follows:

1. High steam line space temperature
2. High steam line flow
3. Low steam line pressure
4. High temperature in the main steam line tunnel (30 minute delay for the HPCI and 60 minute delay for the RCIC)

2. The closure time shall not be less than 3 sec.

TABLE 4.7.1
PENETRATIONS AND SEALS SUBJECT TO TYPE B TESTING

Penetration Number	Identification	Number of Penetrations
X-7A, D	Main Steam Line A, D	4
X-9A, B	Feedwater Line A, B	2
X-11	HPCI Steam Line	1
X-12	Shutdown Cooling Supply	1
X-13A, B	RHR Return to Reactor	2
X-14	Supply to Reactor Water Cleanup	1
X-16A, B	Core Spray to Reactor	2
X-1	Equipment Access Hatch	1
X-3	Drywell Head Flange	1
X-4	Drywell Head Access Hatch	2
X-6	CRD Removal Hatch	1
SLH-A, H	Shear Lug Access Covers	8
X-202A, H & J, K	Vacuum Relief Access Covers	10
X-213A, B	Torus Drains	2
X-200A, B	Torus Manways	2

Bases:

3.7 A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during his period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.25% delta k. A drop of a 1.25% delta k rod does not result in any fuel damage.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable pressure suppression chamber pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 44 psig, which is below the design of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

3.8 LIMITING CONDITIONS FOR OPERATION

3.8 STATION RADIOACTIVE WASTE CONTROL SYSTEMS

Applicability:

Applies to the radioactive effluents from the station.

Objective:

To assume that radioactive material is released to the environment only in a controlled manner, is kept as low as practicable, and is well below the limits of 10 CFR 20.

Specification:

A. Airborne Effluents

1. Radioactive gases released from the plant stack shall be continuously monitored. To accomplish this, as least one plant stack monitoring system, one off-gas monitoring system and an area gamma radiation monitor on the perimeter fence at the stack shall be operable at all times.

2. The activity of noble gases released shall not exceed the rate of 0.22 curie per second.

4.8 SURVEILLANCE REQUIREMENTS

4.8 STATION RADIOACTIVE WASTE CONTROL SYSTEMS

Applicability:

Applies to the periodic monitoring and recording of radioactive effluents.

Objective:

To ascertain that radioactive releases are as low as practicable and well within allowable values.

Specification:

A. Airborne Effluents

1. The plant stack and off-gas system radiation monitoring systems shall be checked daily and calibrated every three months.

2. a. Station records of gross stack release rate of gaseous activity shall be maintained on an hourly basis by one of the following methods:

   (1) Station data logger
   (2) RMS Channel Chart recorder.
   (3) RMS Channel logarithmic ratemeter.

3.8 LIMITING CONDITIONS FOR OPERATION

E. Radioactive Waste Storage

The maximum amount of radioactivity in liquid storage in the the Waste Sample Tanks, the Floor Drain Sample Tanks and the Waste Surge Tank shall not exceed 0.3 curie. If this condition cannot be met, the stored liquid shall be recycled within 24 hours to the Waste Collector Tanks until the condition is met.

F. General

1. Consistent with the principle of keeping levels of radioactive materials in effluents as low as practicable, administrative limits shall be imposed upon plant effluents. Observance of these administrative limits shall require that waste treatment systems be utilized during normal conditions of plant operation in a manner that will:

    a. Reduce the average annual release rate of airborne effluents to not more than 10% of the limit of Specification 3.8.A.2

    b. Reduce the average annual concentration in liquid effluents to not more than 1% of the limits of Specification 3.8.B.2.

2. If it appears that, based on operating experience, these reduced limits will be exceeded, the licensee will exert his best efforts to keep the release of radioactive effluents as low as practicable.

4.8 SURVEILLANCE REQUIREMENTS

E. Radioactive Waste Storage

A sample from each of the Waste Sample Tanks, Floor Drain Sample Tanks, and Waste Surge Tank shall be taken, analyzed and recorded once each 72 hours. Whenever the total activity in a tank exceeds 0.1 curie, a sample shall be taken, analyzed and recorded within 48 hours after each addition to that tank.

F. General

Operating procedures shall be developed and used, and equipment which has been installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used, to keep levels of radioactive material in effluents released to unrestricted areas as low as practicable. Release of radioactive effluents in excess of the administrative limit in specification 3.8.F shal require the review, evaluation and concurrence of the Nuclear Safety Audit and Review Committee.

The environmental monitoring program given in Table 4.8.1 shall be conducted.

6.7 PLANT REPORTING REQUIREMENTS

In addition to reports required by applicable regulations, Vermont Yankee Nuclear Power Corporation shall provide the following information:

A. Events requiring reports within 24 hours (by telephone and telegraph to Region I Compliance Office):

a. Incidents or conditions relating to operation of the Plant which prevented or could have prevented the performance of engineered safety features as described in these Specifications.

b. Any significant variation of measured values from a corresponding predicted value of safety-connected operating parameters occurring during the initial criticality of the Plant.

c. Any abnormal occurrences as specified in the Definitions Section of these Specifications.

d. Incidents or conditions which resulted in a safety limit being exceeded as established in these specifications.

B. Events requiring reports within 10 days (in writing to the Director, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D.C. 20545):

a. In the event a redundant component (or system) is determined to be out of service for periods longer than those specified in other sections, it shall be the subject of a special report. This report shall describe:

1. The nature of the problem and the specific steps to be taken to remedy the situation.

2. An estimate of the time required to return the component (or system) to an operable condition.

3. The amount of component (or system) redundancy remaining of the availability of other system(s) to perform the same function as the inoperable component (or system).

4. Surveillance requirements on the operable component (or system).

b. Any variation of measured values from a corresponding predicted value of safety-connected operating parameters occurring during the initial criticality of the Plant.

c. Incidents or conditions relating to operation of the Plant which prevented the performance of engineered safety features.

6.7.B (cont'd)

d. Any abnormal occurrences as specified in the Definitions Section of these Specifications.

C. Events requiring reports within 30 days (in writing to the Director, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D.C. 20545):

a. Any substantial variance from performance specifications contained in these Specifications or in the Final Safety Analysis Report.

b. Any significant change in transient or accident analysis as described in the Final Safety Analysis Report.

c. Any changes in plant organization as described in 6.1.

d. Observed inadequacies in the implementation of administrative or procedural controls.

The written report, and to the extent possible, the preliminary telephone and telegraph report, should describe, analyze and evaluate safety implications and outline the measures taken to assure that the cause of the condition is determined and to indicate the corrective action, including any significant changes made to procedures and to the quality assurance program, taken to preclude repetition of the occurrence and of similar occurrences involving similar components or systems.

In addition, the written report should relate any failures or degraded performance of systems and components for this incident to similar equipment failures that may have occurred previously at the facility. The evaluation of the safety implications of the incident should consider the cumulative experience obtained from the record of previous failures and malfunctions of the affected systems and components or of similar equipment.

D. Routine Operating Reports (in writing to the Director, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D.C. 20545):

1. A routine operating report shall be submitted monthly. The following information shall be provided:

a. Nuclear

(1) Number of hours the plant was operated.

(2) Number of times the reactor was made critical.

6.7.D (cont'd)

(3) Gross thermal power generated (in MWH).

(4) Operating histogram, showing the thermal power level of the reactor versus time, for the report period.

(5) Equivalent full power hours.

b. Electrical

(1) Gross power generated (in MWH).

(2) Net power generated (in MWH).

(3) Length of time generator was on line (in hours).

c. Shutdowns

(1) Number of scrams and shutdowns.

(2) Duration of downtime (in hours).

(3) Reasons for outage.

d. Maintenance

(1) Nature of the maintenance; e.g., routine, emergency, preventive or corrective.

(2) The effect, if any, on the safe operation of the reactor.

(3) The cause of any malfunction for which corrective maintenance was required.

(4) The results of any such malfunctions.

(5) Corrective and preventive action taken to preclude recurrence.

(6) Time required for completion.

6.7.D (cont'd)

2. An effluent release report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year of operation specifying total quantities of radioactive material released to unrestricted areas in liquid and gaseous effluents during the previous six months and such other information on releases as may be required to estimate exposures to the public resulting from effluent releases. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operational occurrences, the report shall cover this specifically. The following information summarized on a monthly basis, shall be provided.

   a. Environmental Monitoring

      (1) Descriptive material covering the offsite environmental surveys performed during the reporting period including information on:

         (a) The number and types of samples taken; e.g., air, river, bottom, surface water soil, fish.

         (b) The number and types of measurements made; e.g., dosimetry.

         (c) Locations of the sample points and monitoring stations.

         (d) The frequency of the surveys.

         (e) A summary of survey results.

      (2) If a particular sample or measurement indicates statistically significant levels of radioactivity above established or concurrent backgrounds, the following information shall be provided:

         (a) The type of analysis performed; e.g., alpha, beta, gamma and/or isotopic.

         (b) The minimum sensitivity of the monitoring system.

         (c) The measured radiation level or sample concentration.

         (d) The specific times when samples were taken and measurements were made.

         (e) An estimate of the likely resultant exposure to the public.

6.7.D (cont'd)

b. Radioactive Liquid Wastes

(1) Gross Radioactivity (beta, gamma)

(a) Total release, curies.

(b) Average concentration, µCi/ml, at the outfall of the discharge structure.

(c) Maximum concentration, µCi/ml, at the outfall of the discharge structure, including the date, time and duration of the release.

(2) Tritium

(a) Total release, curies.

(b) Average concentration, µCi/ml, at the outfall of the discharge structure.

(3) Dissolved Noble Gases

(a) Total release, curies.

(b) Average concentration, µCi/ml, at the outfall of the discharge structure.

(4) Gross Alpha Radioactivity

(a) Total release, curies.

(b) Average concentration, µCi/ml, at the outfall of the discharge structure.

(5) Volume, liters, of Liquid Wastes Released.

(6) Volume, liters, of Dilution Water.

(7) Results of individual Radionuclide Identification and the total release, curies, of each identified nuclide.

(8) The percent of the limit specified in 3.8.B.2 for total activity or a specified nuclide.

6.7.D (cont'd)

c. Radioactive Airborne Releases

(1) Noble and Activation Gases

(a) Total release, curies.

(b) Maximum release rate, μCi/sec, during any consecutive 24 hours and for any one-hour period, including the date, time and the meteorological diffusion conditions.

(2) Iodine

(a) Total release, curies.

(3) Particulates

(a) Total gross radioactivity (beta, gamma) release, curies.

(b) Total gross alpha release, curies.

(4) Tritium

(a) Total release, curies.

(5) Results of Individual Radionuclide Identification and the total release, curies, of each identified nuclide.

(6) The percent of the limit specified in 3.8.A.

d. Solid Radioactive Waste

(1) Total volume, cubic feet.

(2) Gross activity, curies.

(3) Dates and disposition of material, if shipped offsite.

E. Special Reports

1. A comprehensive report presenting the results of the initial pre-operational startup, power ascension and full power test programs shall be submitted within one year of the commercial date.

6.7.E (cont'd)

2. Reports on the following areas shall be submitted as noted:

	Area	Reference	Submittal Date
a.	Containment Leak Rate Testing (1)	4.7	Upon completion of each test
b.	In-service Inspection Evaluation	4.6	As per Section XI ASME Boiler Codes

(1) Each integrated leak rate test shall be the subject of a summary technical report, which includes a schematic arrangement of the leakage measurement system, the pressure, temperature, and humidity instrumentation employed including their sensitivity, the test procedures, test results in graphical and tabular form and the analysis and interpretation of leakage rate results in meeting the allowable leakage rates specified in the license. Summaries of local leak test results shall be included in the same report to permit evaluation of local leak testing as compared to integrated leak rate testing.

c. A report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year of operation specifying total quantities of radioactive material released to unrestricted areas in liquid and gaseous effluents during the previous six months and such other information on releases as may be required to estimate exposures to the public resulting from effluent releases. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operational occurrences, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the Licensee or others, the Commission may from time to time require the Licensee to take such action as the Commission deems appropriate.