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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-338

NORTH ANNA POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. NPF-4

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated March 16, 1984 and revised November 21, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.D.(2) of Facility Operating License No. NPF-4 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 7 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 1, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 63

TO FACILITY OPERATING LICENSE NO. NPF-4

DOCKET NO. 50-338

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Replace
Ia	Ia
XX	XX
1-5	1-5
3/4 2-1	3/4 2-1
3/4 2-7	3/4 2-7
3/4 2-17	3/4 2-17
3/4 3-52	3/4 3-52
3/4 3-56	3/4 3-56
3/4 3-58	3/4 3-58
3/4 3-65	3/4 3-65
3/4 4-13	3/4 4-13
3/4 7-75	3/4 7-75
3/4 7-79	3/4 7-79
3/4 7-80	3/4 7-80
3/4 7-81	3/4 7-81
3/4 11-1	3/4 11-1
3/4 11-8	3/4 11-8
3/4 11-18	3/4 11-18
3/4 12-1	3/4 12-1
3/4 12-2	3/4 12-2
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3/4 12-12	3/4 12-12
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3/4 12-14	3/4 12-14
B 3/4 4-3	B 3/4 4-3
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DEFINITIONS

SECTION	PAGE
1.0 DEFINITIONS	

INDEX (Cont'd)

ADMINISTRATIVE CONTROLS

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be tha qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 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 component at the beginning of each subinterval.

THERMAL POWER

1.32 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a $\pm$ 5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the $\pm$ 5% target band about the target flux difference and with THERMAL POWER:

1. Above P_f% of RATED THERMAL POWER, within 15 minutes, where $P_f=(0.9xP_m)$; the value for P_m is established in the Core Surveillance Report per Technical Specification 6.9.1.7.

    a) Either restore the indicated AFD to within the target band limits, or

    b) Reduce THERMAL POWER to less than P_f% of RATED THERMAL POWER.

2. Between 50% and P_f% of RATED THERMAL POWER:

    a) POWER OPERATION may continue provided:

        1) The indicated AFD has not been outside of the $\pm$5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

        2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

    b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1. provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above P_f% of RATED THERMAL POWER unless the indicated AFD is within the ±5% target band and ACTION 2.a.1, above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ±5% target band when at least 2 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ±5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

(b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for Rated Thermal Power (F_{xy}RTP) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes, in a Core Surveillance Report per Technical Specification 6.9.1.7.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ±2%, 32.1 ±2%, 46.4±2%, 60.6±2% and 74.9±2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

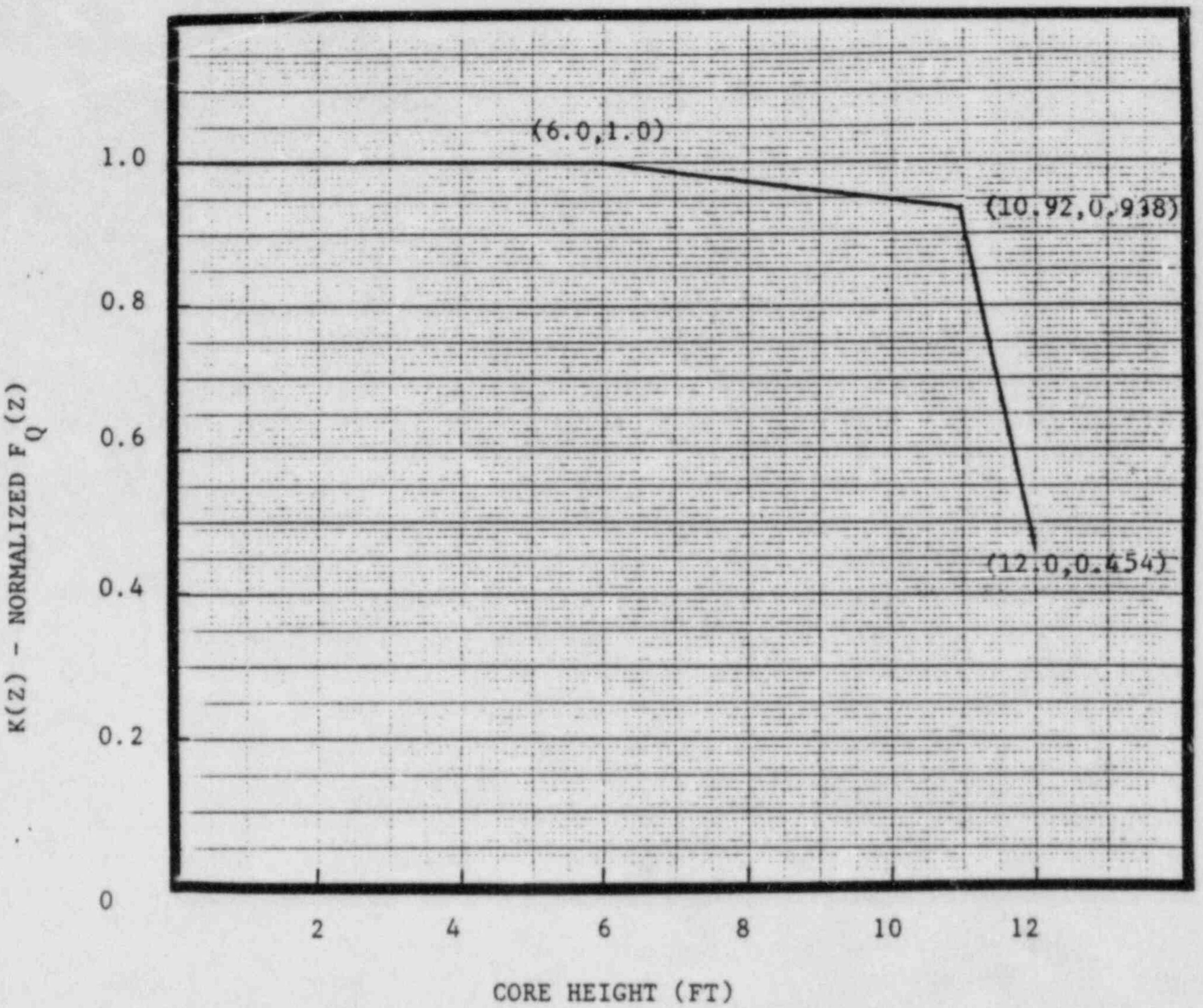

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

e. σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of R_j, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2 \right]^{1/2}}{\overline{R}_j}$$

f. The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system, respectively.

g. The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 ABOVE P_m% of RATED THERMAL POWER#, where the value for P_m is established in the Core Surveillance Report per Technical Specification 6.9.1.7.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by $\leq$ 4 percent, reduce THERMAL POWER one percent for every percent by which the $F_j(Z)$ factor exceeds its limit within 15 minutes and within the next two hours either reduce the $F_j(Z)$ factor to within its limit or reduce THERMAL POWER to P_m% or less of RATED THERMAL POWER.

b. With a $F_j(Z)$ factor exceeding $[F_j(Z)]_S$ by > 4 percent, reduce THERMAL POWER to P_m% or less of RATED THERMAL POWER within 15 minutes.

The APDMS may be out of service when surveillance for determining power distribution maps is being performed.

SURVEILLANCE REQUIREMENTS

4.2.6.1 $F_j(Z)$ shall be determined to be within its limit by:

a. Either using the APDMS to monitor the thimbles required per Specification 3.3.3.8 at the following frequencies.

1. At least once per 8 hours, and

2. Immediately and at intervals of 10, 30, 60, 90, 120, 240 and 480 minutes following:

   a) Increasing the THERMAL POWER above P_m% of RATED THERMAL POWER, or

   b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

b. Or using the movable incore detectors at the following frequencies when the APDMS is inoperable:

1. At least once per 8 hours, and

2. At intervals of 30, 60, 90, 120, 240 and 480 minutes following:

   a) Increasing the THERMAL POWER above P_m% of RATED THERMAL POWER, or

   b) Movement of control bank "D" more than an accumulated total of 5 steps in any one direction.

4.2.6.2 When the movable incore detectors are used to monitor $F_j(Z)$, at least 2 thimbles shall be monitored and an $F_j(Z)$ accuracy equivalent to that obtained from the APDMS shall be maintained.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Inlet Temperature-T_{hot} (wide range)	M	R
3. Reactor Coolant Inlet Temperature-T_{cold} (wide range)	M	R
4. Reactor Coolant Pressure-Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level-Narrow Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Boric Acid Tank Solution Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position Indicator	M	R

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11 inoperable:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The NFPA Code 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits between the local panels in Specification 4.3.3.7.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.8.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

b. At least once per 18 months, by performance of a CHANNEL CALIBRATION.

LOOSE PARTS MONITORING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.3.3.9 The loose parts monitoring system instrumentation identified in Table 3.3-12 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION

If all channels of one or more collection regions are inoperable, restore the instrument(s) to OPERABLE status within 30 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channels to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the loose parts monitoring system identified in Table 3.3-12 shall be demonstrated OPERABLE by the performance of:

a. A CHANNEL CHECK at least once per 24 hours.

b. A CHANNEL FUNCTIONAL TEST at least once per 31 days.

c. A CHANNEL CALIBRATION at least once per 18 months.

LOOSE PARTS MONITORING SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.3.3.9 The loose parts monitoring system instrumentation identified in Table 3.3-12 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION

If all channels of one or more collection regions are inoperable, restore the instrument(s) to OPERABLE status within 30 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channels to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each channel of the loose parts monitoring system identified in Table 3.3-12 shall be demonstrated OPERABLE by the performance of:

a. A CHANNEL CHECK at least once per 24 hours.

b. A CHANNEL FUNCTIONAL TEST at least once per 31 days.

c. A CHANNEL CALIBRATION at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

a. If the absolute value of $\frac{R_{ij}-\overline{R}_j}{\overline{R}_j}$ is greater than $2\sigma_j$, another map shall be completed to verify the new $\overline{R}_j$. If the second map shows the first to be in error, the first map shall be disregarded. If the second map confirms the new $\overline{R}_j$, four more maps (including rodded configurations allowed by the insertion limits) will be completed so that a new $\overline{R}_j$ and σ_j can be defined from the six new maps.

4.3.3.8.2 The APDMS shall be demonstrated OPERABLE:

a. By performance of a CHANNEL FUNCTIONAL TEST within 7 days prior to its use and at least once per 31 days thereafter when used for monitoring $F_j(Z)$.

b. At least once per 18 months, by performance of a CHANNEL CALIBRATION.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

Table 3.3-13

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	26
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	1	26
b. Circulating Water System Effluent Line	1	29
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
a. Clarifier Effluent Line	1	26
4. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	27

TABLE 4.3-13 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Instrument controls not set in operate mode.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-14 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-14.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-14. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-14.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.1 The fire suppression water system shall be OPERABLE with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies from the North Anna Reservoir and the Service Water Reservoir, and

c. An OPERABLE flow path capable of taking suction from the North Anna Reservoir and the Service Water Reservoir and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the valve at each hose standpipe as required to be OPERABLE per Specification 3.7.14.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. Submit a Special Report in accordance with Specification 6.9.2:

      a) By telephone within 24 hours,

      b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

LIMITING CONDITION FOR OPERATION (Continued)

c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.14.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. By verifying the contained water supply volumes pursuant to Specification 4.7.5.1.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

d. By performance of a system flush as necessary to maintain the system water chemistry within acceptable limits.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

1. Verifying that each automatic valve in the flow path actuates to its correct position,

2. Verifying that each pump develops at least 2500 gpm at a system head of $\geq$ 250 feet for 1-FP-P-1 and 187 feet for 1-FP-P-2.

3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

4. Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure $\geq$ 80 psig in the main fire loop.

LOW PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following low pressure CO_2 systems shall be OPERABLE with a minimum of 3.5 tons in the storage tank at a minimum pressure of 275 psig.

a. Cable tunnels and vaults
b. Charcoal filters
c. Emergency diesel generator rooms

APPLICABILITY: Whenever equipment in the low pressure CO_2 protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required low pressure CO_2 systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying CO_2 storage tank level and pressure, and

b. At least once per 18 months by verifying:

   1. The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

   2. Flow from each nozzle during a "Puff Test."

HIGH PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.3 The following high pressure CO_2 systems shall be OPERABLE with the storage tanks having at least 90% of full charge weight.

a. Fuel oil pump rooms

APPLICABILITY: Whenever equipment in the high pressure CO_2 protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required high pressure CO_2 systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying CO_2 storage tank weight.

b. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

a. Control Room

APPLICABILITY: Whenever equipment in the Halon protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying Halon storage tank weight and pressure.

b. At least once per 18 months by:

   1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

   2. Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.14.5. The fire hose stations shown in Table 3.7-7 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-7 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.5 Each of the fire hose stations shown in Table 3.7-7 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) 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 shall be limited to 2×10^{-4} microcuries/ml.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (mCi/ml)
A. Batch Releases[b,g]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
			Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Continuous[f]	M Composite[f]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank**

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per week when radioactive materials are being added to the tank.

*This is a shared system with Unit 2.

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

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to the critical organ.*

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined continuously to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

*The critical organ is defined in the ODCM.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to $\leq$ 25,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 uCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 SOLIDIFICATION shall be conducted in accordance with a PROCESS CONTROL PROGRAM.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternate SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 4.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 4.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.8, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. 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 4.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 4.12-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.12-2 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 3.11.1.2, 3.11.2.2 and 3.11.2.3. 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 Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.12-1, 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. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.12-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 4.12-1, the detection capabilities required by Table 4.12-3, and the guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, "October 1978, and to Radiological Assessment Branch Technical Positions, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

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

[e]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[f]If milk sampling cannot be performed, use item 4.c.

TABLE 4.12-3 (Continued)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

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

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

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical valves of E, V, Y and Δt should be used in the calculation.

TABLE 4.12-2 (Continued)

TABLE NOTATION

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. Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 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.

APPLICABILITY: At all times.

ACTION:

a. 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 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 25 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, 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. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 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 Operating Report pursuant to Specification 6.9.1.8.

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.12-1.4c shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials (which contain nuclides produced at nuclear power stations) supplied as part of an Interlaboratory Comparison Program, described in the ODCM, that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be maintained as described in the ODCM.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (ATV) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Section 50.72 to 10 CFR Part 50 with a followup report pursuant to Section 50.73 to 10 CFR Part 50. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are generally consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 30 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDRY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1 - 1971 for comparable positions and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, except for (1) the Supervisor - Health Physics who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 The Station Manager is responsible for ensuring that retraining and replacement training programs for the facility staff are maintained and that such programs meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1 - 1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the SES.

6.5 REVIEW AND AUDIT

6.5.1 STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC)

FUNCTION

6.5.1.1 The SNSOC shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SNSOC shall be composed of the :

Chairman:	Station Manager
Vice Chairman:	Assistant Station Manager
Member:	Superintendent-Operations
Member:	Superintendent-Maintenance
Member:	Superintendent-Technical Services
Member:	Supervisor-Health Physics

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the SNSOC Chairman to serve on a temporary basis; however, no more than one alternate shall participate as a voting member in SNSOC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The SNSOC shall meet at least once per calendar month and as convened by the SNSOC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all procedures required by Specifications 6.8.1, 6.8.2 and 6.8.3 and changes tnereto, 2) all programs required by Specification 6.8.4 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

f. Review of all REPORTABLE EVENTS and Special Reports.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

d. Violations, REPORTABLE EVENTS and Special Reports such as:

1. Violations of applicable codes, resulations, orders, Technical Specifications, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations from normal or expected performance of station safety-related structures, systems, or components; and

3. All REPORTABLE EVENTS submitted in accordance with Section 50.73 to 10 CFR Part 50 and Special Reports required by Specification 6.9.2.

Review of events covered under this paragraph shall include the results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. The Quality Assurance Department audit program at least once per 12 months and audit reports.

f. Any other matter involving safe operation of the nuclear power stations which is referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee.

g. Reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

AUTHORITY

6.5.2.9 The Director-Safety Evaluation and Control shall report to and advise the Manager-Nuclear Technical Services, who shall advise the Vice President-Nuclear Operations on those areas of responsibility specified in Section 6.5.2.7.

RECORDS

6.5.2.10 Records of SEC activities required by Section 6.5.2.7 shall be prepared and maintained in the SEC files and a summary shall be disseminated as indicated below each calendar month.

1. Vice President-Nuclear Operations
2. Nuclear Power Station Managers
3. Manager-Nuclear Operations and Maintenance
4. Manager-Nuclear Technical Services
5. Manager-Quality Assurance, Operations
6. Others that the Director-Safety Evaluation and Control may designate.

6.5.3 QUALITY ASSURANCE DEPARTMENT

FUNCTION

6.5.3.1 The Quality Assurance Department shall function to audit station activities. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Station Emergency Plan and implementing procedures at least once per 12 months.

f. The Station Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the Executive Manager-Quality Assurance or the Senior Vice President-Power Operations.

h. The Station Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

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

AUTHORITY

6.5.3.2 The Quality Assurance Department shall report to and advise the Executive Manager-Quality Assurance, who shall advise the Senior Vice President-Power Operations on those areas of responsibility specified in Section 6.5.3.1.

RECORDS

6.5.3.3. Records of the Quality Assurance Department audits shall be prepared and maintained in the department files. Audit reports shall be disseminated as indicated below:

1. Nuclear Power Station Manager
2. Manager-Nuclear Operations and Maintenance
3. Manager-Nuclear Technical Services
4. Manager-Quality Assurance, Operations
5. Director-Quality Assurance, Nuclear Operations
6. Director-Safety Evaluation and Control
7. Supervisor of area audited
8. Nuclear Power Station Resident Quality Control Engineer

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Director-Safety Evaluation and Control and the Manager - Nuclear Operations and Maintenance.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

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

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

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CORE SURVEILLANCE REPORT

6.9.1.7 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) in all core planes containing Bank "D" control rods and in all unrodded core planes, the surveillance power level, P_m, for Technical Specifications 3.2.1 and 3.2.6, and the F_Q flyspeck basis as determined using the definitions and methodology in WCAP 8385 and Westinghouse letter to NRC dated April 6, 1978, Serial No. NS-CE-1749 shall be provided to the Regional Administrator, Region II, with a copy to:

Director, Office of Nuclear Reactor Regulation
Attention: Chief, Core Performance Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

at least 60 days prior to cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Core Surveillance Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter.

Any additional information needed to support the F_{xy}^{RTP} and P_m submittal will be by request from the NRC and need not be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate) with preoperational studies, 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 land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include 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 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 individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1 and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.9 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report shall be submitted within 60 days after January 1 of each year. This report shall include an assessment of the radiation doses to the maximum exposed MEMBERS OF THE PUBLIC due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual Meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Concurrent meteorological conditions or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

If the dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specifications 3.11.1.2a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed MEMBER OF THE PUBLIC shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations.

The Radioactive Effluent Release Reports shall include a list of unplanned releases as required to be reported in Section 50.73 to 10 CFR Part 50 from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirement of the applicable reference specification:

a. Inservice Inspection Program Reviews shall be reported within 90 days of completion. Specification 4.0.5.

b. ECCS Actuation shall be reported within 90 days of the occurrence. The report shall describe the circumstances of the actuation and the total accumulated cycles to date. Specifications 3.5.2 and 3.5.3.

c. With Seismic Monitoring Instrumentation inoperable for more than 30 days, submit a special report within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status. Specification 3.3.3.3.

d. For all seismic events actuating a seismic monitoring instrument, submit a special report within 10 days describing the magnitude, frequency spectrum and resultant effects upon features important to safety. Specification 4.3.3.3.2.

e. With Meteorological Instrumentation inoperable for more than 7 days, submit a special report within the next 10 days, outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status. Specification 3.3.3.4.

f. With the primary coolant specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, a specific activity analysis shall be included in the Special Report. The information requested in Specification 3.4.8 shall also be included in that report.

g. With sealed source or fission detector leakage tests revealing the presence of $\geq$ 0.005 microcuries of removable contamination submit a special report on an annual basis outlining the corrective actions taken to prevent the spread of contamination. Specification 4.7.11.1.3.

h. With the MTC more positive than 0 $\Delta k/k/^{o}F$ submit a special report within the next 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. Specification 3.1.1.4.

i. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.

j. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking, and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.

k. Inoperable Fire Detection Instrumentation, Specification 3.3.3.7.

l. Inoperable Fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2, 3.7.14.3, 3.7.14.4 and 3.7.14.5.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS and Special Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

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

VIRGINIA ELECTRIC AND POWER COMPANY

OLD DOMINION ELECTRIC COOPERATIVE

DOCKET NO. 50-339

NORTH ANNA POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 47
License No. NPF-7

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

   A. The application for amendment by Virginia Electric and Power Company, et al., (the licensee) dated March 16, 1984 and revised November 21, 1984, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-7 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 47 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 7 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 1, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 47

TO FACILITY OPERATING LICENSE NO. NPF-7

DOCKET NO. 50-339

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Replace
Ia	Ia
XVII	XVII
1-5	1-5
3/4 1-5	3/4 1-5
3/4 2-1	3/4 2-1
3/4 2-7	3/4 2-7
3/4 2-18	3/4 2-18
3/4 3-49	3/4 3-49
3/4 3-53	3/4 3-53
3/4 3-60	3/4 3-60
3/4 4-13	3/4 4-13
3/4 6-9	3/4 6-9
3/4 7-59	3/4 7-59
3/4 7-63	3/4 7-63
3/4 7-65	3/4 7-65
3/4 7-67	3/4 7-67
3/4 7-68	3/4 7-68
3/4 7-70	3/4 7-70
3/4 8-6	3/4 8-6
3/4 11-1	3/4 11-1
3/4 11-8	3/4 11-8
3/4 11-18	3/4 11-18
3/4 12-1	3/4 12-1
3/4 12-2	3/4 12-2
3/4 12-7	3/4 12-7
3/4 12-11	3/4 12-11
3/4 12-12	3/4 12-12
3/4 12-13	3/4 12-13
3/4 12-14	3/4 12-14
B 3/4 4-3	B 3/4 4-3
6-7	6-7
6-10	6-10
6-13	6-13
6-14b	6-14b
6-16	6-16
6-17	6-17
6-18	6-18
6-19	6-19
6-20	6-20
6-21	6-21
6-22	6-22

DEFINITIONS

SECTION PAGE

1.0 DEFINITIONS

ADMINISTRATIVE CONTROLS

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.26 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet wastes into a solid form that meets shipping and burial ground requirements.

SOURCE CHECK

1.30 A SOURCE CHECK shall be tha qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 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 component at the beginning of each subinterval.

THERMAL POWER

1.32 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY where access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 delta $k/k/^{o}F$ for the all rods withdrawn, beginning of core life, hot zero THERMAL POWER condition, and

b. Less negative than -4.0×10^{-4} delta $k/k/^{o}F$ for the all rods withdrawn, end of core life at RATED THERMAL POWER.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#.
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#.

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a. above, operations in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 delta $k/k/^{o}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. The control rods are maintained within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of 3.1.1.4.b. above, be in HOT SHUTDOWN within 12 hours.

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*With K_{eff} greater than or equal to 1.0

#See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL Limit of Specification 3.1.1.4.a. above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -3.1×10^{-4} delta k/k/°F (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicated the MTC is more negative than -3.1×10^{-4} delta k/k/°F, the MTC shall be remeasured, and compared to the EOL MTC limit of specification 3.1.1.4.b., at least once per 14 EFPD during the remainder of the fuel cycle.

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ± 5% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 5% target band about the target flux difference and with THERMAL POWER:

   1. Above P_f% of RATED THERMAL POWER, within 15 minutes, where $P_f=(0.9xP_m)$; the value for P_m is established in the Core Surveillance Report per Technical Specification 6.9.1.7.

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than P_f% of RATED THERMAL POWER.

   2. Between 50% and P_f% of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

      b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1. provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours of operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2.

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above P_f% of RATED THERMAL POWER unless the indicated AFD is within the ±5% target band and ACTION 2.a.1, above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ±5% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ±5% target band when at least 2 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the ±5% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

(b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for Rated Thermal Power (F_{xy}RTP) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes, in a Core Surveillance Report per Technical Specification 6.9.1.7.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.

2. Upper core region from 85 to 100%, inclusive.

3. Grid plane regions at 17.8 ±2%, 32.1 ±2%, 46.4±2%, 60.6±2% and 74.9±2%, inclusive (17 x 17 fuel elements).

4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L} the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determination, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

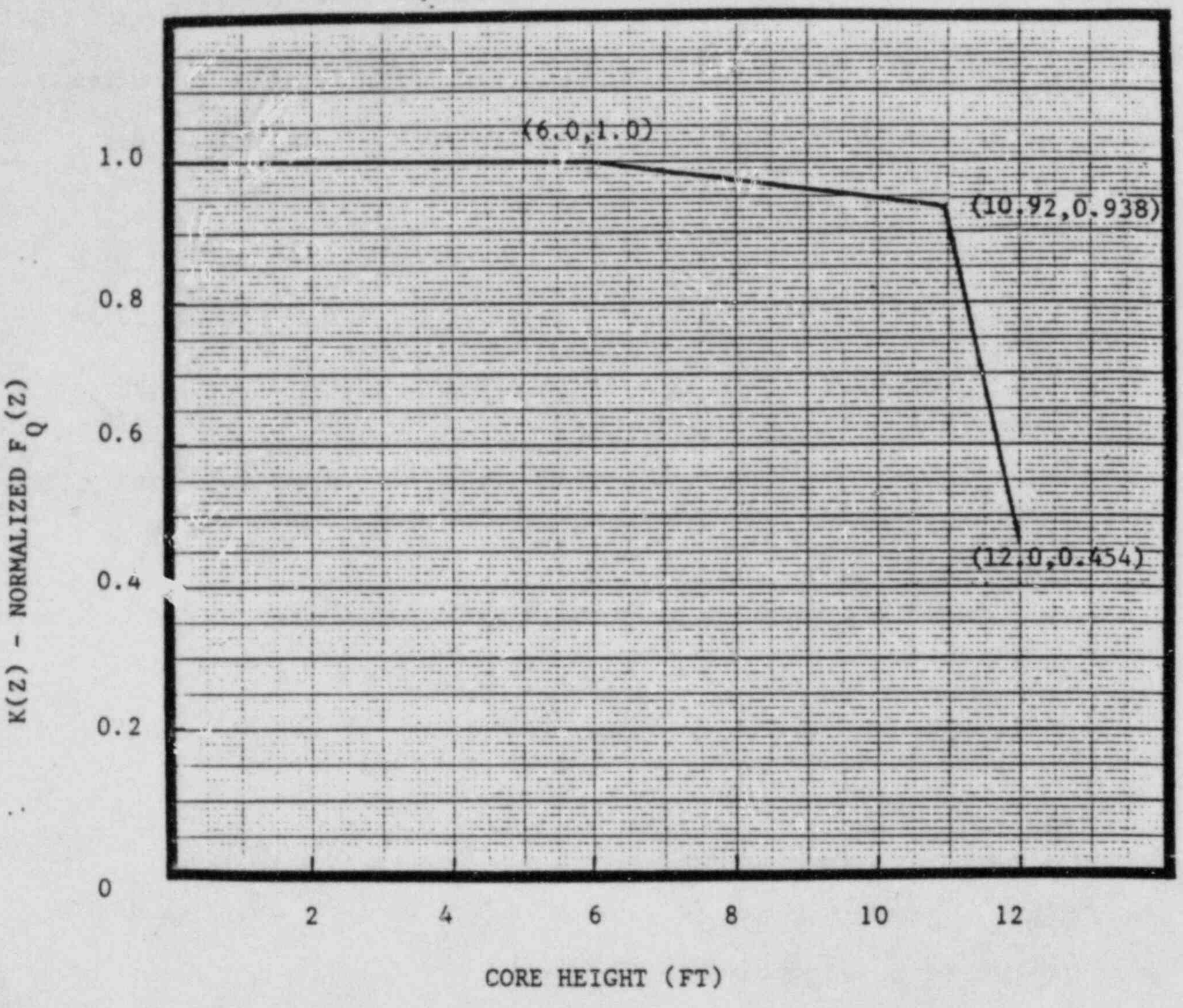

Figure 3.2-2 K(Z) - Normalized $F_Q(Z)$ as a Function of Core Height

AXIAL POWER DISTRIBUTION

LIMITING CONDITION FOR OPERATION

3.2.6 The axial power distribution shall be limited by the following relationship:

$$[F_j(Z)]_S = \frac{[2.20]\ [K(Z)]}{(\overline{R}_j)(P_L)(1.03)(1 + \sigma_j)(1.07)}$$

Where:

a. $F_j(Z)$ is the normalized axial power distribution from thimble j at core elevation Z.

b. P_L is the fraction of RATED THERMAL POWER.

c. K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

d. $\overline{R}_j$, for thimble j, is determined from at least n=6 incore flux maps covering the full configuration of permissible rod patterns above P_m% of RATED THERMAL POWER in accordance with:

$$\overline{R}_j = \frac{1}{n} \sum_{i=1}^{n} R_{ij}$$

Where:

$$R_{ij} = \frac{F_{Qi}^{Meas}}{[F_{ij}(Z)]_{Max}}$$

and $[F_{ij}(Z)]_{Max}$ is the maximum value of the normalized axial distribution at elevation Z from thimble j in map i

LIMITING CONDITION FOR OPERATION (Continued)

which had a measured peaking factor without uncertainties or densification allowance of F_Q^{Meas}.

e. σ_j is the standard deviation associated with thimble j, expressed as a fraction or percentage of $\overline{R}_j$, and is derived from n flux maps from the relationship below, or 0.02, (2%) whichever is greater.

$$\sigma_j = \frac{\left[\frac{1}{n-1} \sum_{i=1}^{n} (\overline{R}_j - R_{ij})^2\right]^{1/2}}{\overline{R}_j}$$

f. The factor 1.07 is comprised of 1.02 and 1.05 to account for the axial power distribution instrumentation accuracy and the measurement uncertainty associated with F_Q using the movable detector system, respectively,

g. The factor 1.03 is the engineering uncertainty factor.

APPLICABILITY: MODE 1 ABOVE P_m% of RATED THERMAL POWER#, where the value for P_m is established in the Core Surveillance Report per Technical Specification 6.9.1.7.

ACTION:

a. With a $F_j(Z)$ factor exceeding $[F_j(Z)_S]$ by less than or equal to 4 percent, reduce THERMAL POWER one percent for every percent by which

#The APDMS may be out of service when surveillance for determining power distribution maps being performed.

INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11 inoperable:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The NFPA Code 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits between the local panels in Specification 4.3.3.7.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTATION

INSTRUMENT LOCATION	MINIMUM DETECTORS REQUIRED HEAT	SMOKE
1. Reactor Containment		
a. Reactor Coolant Pumps	1/pump*	
b. Residual Heat Removal Pump Area	3	
c. Cable Penetration Area	7	8
d. Recirculation Air System		2
2. Control Room		
a. Under Floor - Loop 1	2	2
b. Under Floor - Loop 2	2	
c. Normal Air Supply#		1
d. Emergency Air Supply		1
e. Ceiling Area		10
f. Return Air Duct		1
3. Cable Spreading Room	3	4
4. Primary Cable Vault and Tunnel	2	3
5. Service Building Cable Vault and Tunnel	5	4
6. Emergency Switchgear Rooms		
a. Emergency Air Supply		1
b. Emergency Switchgear and Air Conditioning Rooms		7
7. Station Battery Room		1/room
8. Diesel Generators	2/room	
9. Fuel Oil Pump House#		
a. Room 1	1	1
b. Room 2	1	1
c. Motor Control Center Room		1
10. Rod Control Equipment and Motor Control Center Room (Elevation 280.0)		2
11. Auxiliary Building		
a. Charging Pump Cubicles		1/cubicle
b. Exhaust Duct (Northeast-Cubicles)#		1
c. Exhaust Duct (South Central-Cubicles)#		1
d. Exhaust Duct (Northwest-Cubicles)#		1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	26
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	1	26
b. Circulating Water System Effluent Line	1	29
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
a. Clarifier Effluent Line	1	26
4. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	27

TABLE 4.3-12 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Instrument controls not set in operate mode.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspection.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported on an annual basis for the period in which this inspection was completed. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 require prompt notification of the Commission pursuant to Section 50.72 to 10 CFR Part 50. A Licensee Event Report shall be submitted pursuant to Section 50.73 to 10 CFR Part 50 and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.6.2 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.2. This report shall include a description of the condition of the concrete and liner, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT QUENCH SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment quench spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one containment quench spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment quench spray subsystem shall be demonstrated OPERABLE:

- a. At least once per 31 days by:
  1. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
  2. Verifying the temperature of the borated water in the refueling water storage tank is within the limits shown on Figure 3.6-1.
- b. Verifying that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 123 psig when tested pursuant to Specification 4.0.5.
- c. At least once per 18 months during shutdown, by:
  1. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--high-high signal.
  2. Verifying that each spray pump starts automatically on a Containment Pressure--high-high signal.
- d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

3/4.7.14 FIRE SUPPRESSION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.1 The fire suppression water system shall be OPERABLE with:

a. Two high pressure pumps, each with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies from the North Anna Reservoir and the Service Water Reservoir, and

c. An OPERABLE flow path capable of taking suction from the North Anna Reservoir and the Service Water Reservoir and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves and the valve at each hose standpipe as required to be OPERABLE per Specification 3.7.14.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to provide the loss of redundancy in this system. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. Submit a Special Report in accordance with Specification 6.9.2:

      a) By telephone within 24 hours,

      b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

LIMITING CONDITION FOR OPERATION (Continued)

c. In writing within 14 days following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.7.14.1.1 The fire suppression water system shall be demonstrated OPERABLE:

a. By verifying the contained water supply volumes pursuant to Specification 4.7.5.1.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

d. By performance of a system flush as necessary to maintain the system water chemistry within acceptable limits.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

    1. Verifying that each automatic valve in the flow path actuates to its correct position,

    2. Verifying that each pump develops at least 2500 gpm at a system head of greater than or equal to 250 feet for 1-FP-P-1 and greater than or equal to 187 feet for 1-FP-P-2.

    3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

    4. Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 80 psig in the main fire loop.

LOW PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following low pressure CO_2 systems shall be OPERABLE with a minimum of 3.5 tons in the storage tank (common to Units 1 and 2) at a minimum pressure of 275 psig:

a. Cable tunnels and vaults

b. Charcoal filters

c. Emergency diesel generator rooms

APPLICABILITY: Whenever equipment protected by the low pressure CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more of the above required low pressure CO_2 systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.2. Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by verifying CO_2 storage tank level and pressure.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its connect position.

c. At least once per 18 months by verifying:

   1. The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

   2. Flow from each nozzle during a "Puff Test."

PLANT SYSTEMS

HIGH PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.3 The following high pressure CO_2 systems shall be OPERABLE with the storage tanks having at least 90% of full charge weight:

a. Fuel oil pump rooms

APPLICABILITY: Whenever equipment protected by the high pressure CO_2 system is required to be OPERABLE.

ACTION:

a. With one or more the above required high pressure CO_2 systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.3 Each of the above required high pressure CO_2 systems shall be demonstrated OPERABLE:

HIGH PRESSURE CO_2 SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 6 months by verifying CO_2 storage tank weight.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

c. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure:

a. Control Room

APPLICABILITY: Whenever equipment protected by the Halon system is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon systems inoperable, within one hour, establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.4 Each of the above required Halon systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight and pressure.

c. At least once per 18 months by:

1. Verifying the system, including associated ventilation dampers, actuates manually and automatically, upon receipt of a simulated test signal, and

2. Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITIONS FOR OPERATION

3.7.14.5 The fire hose stations shown in Table 3.7-7 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-7 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the fire hose station to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.14.5 Each of the fire hose stations shown in Table 3.7-7 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at that hose station.

SURVEILLANCE REQUIREMENTS

b. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system and verifying that the automatic valves in the flow path actuate to their correct positions.

2. By inspection of spray headers to verify their integrity.

3. By inspection of each nozzle to verify no blockage.

c. At least once per 3 years by an air flow test of the open head spray and/or sprinkler system.

3/4.7.15 PENETRATION FIRE BARRIERS

LIMITING CONDITIONS FOR OPERATION

3.7.15 All fire barrier penetrations (including cable penetration barriers, firedoors and fire dampers), in fire zone boundaries, protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour, either establish a continuous fire watch on at least one side of the affected penetration, or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.15 Each of the above required penetration fire barriers shall be verified to be functional:

a. At least once per 18 months, by a visual inspection, and

b. Prior to declaring a penetration fire barrier functional following repairs or maintenance by a visual inspection of the affected penetration fire barrier(s).

7. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to 3025 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 2750 kw. Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2.c.4.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3000 kw.

9. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,
   b) Transfer its loads to the offsite power source, and
   c) Proceed through its shutdown sequence.

10. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

   a) Remote Local Selection Switch
   b) Emergency Stop Switch

d. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

4.8.1.1.3 Each diesel generator 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

   2. The pilot cell specific gravity, corrected to 77°F and full electrolyte level, is greater than or equal to 1.200,

   3. The pilot cell voltage is greater than or equal to 2.08 volts, and

SURVEILLANCE REQUIREMENTS (Continued)

4. The overall battery voltage is greater than or equal to 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is greater than or equal to 2.08 volts under float charge and has not decreased more than 0.05 volts from the value observed during the previous test,

2. The specific gravity, corrected to 77°F and full electrolyte level, of each connected cell is greater than or equal to 1.200 and has not decreased more than 0.08 from the value observed during the previous test, and

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

c. At least once per 18 months by verifying that:

1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

2. The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material.

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 0.01 ohms.

4. The battery charger will supply at least ten amperes at 125 volts for at least 4 hours.

d. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-3) 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 shall be limited to 2×10^{-4} microcuries/ml.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (mCi/ml)
A. Batch Releases[b,g]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P Each Batch	M Composite[d]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[f]	$5x10^{-7}$
			I-131	$1x10^{-6}$
			Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous[f]	M Composite[f]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank**

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per week when radioactive materials are being added to the tank.

*This is a shared system with Unit 1.

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

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for tritium, and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to the critical organ.*

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined continuously to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

*The critical organ is defined in the ODCM.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ≤ 25,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is ≤ 1.0 uCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 uCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 SOLIDIFICATION shall be conducted in accordance with a PROCESS CONTROL PROGRAM.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions.

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternate SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 4.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 4.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.8, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. 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 4.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 4.12-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.12-2 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 3.11.1.2, 3.11.2.2 and 3.11.2.3. 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 Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.12-1, 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. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.12-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 4.12-1, the detection capabilities required by Table 4.12-3, and the guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, "October 1978, and to Radiological Assessment Branch Technical Positions, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the cause of the unavailability of samples for that pathway and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

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

[e]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[f]If milk sampling cannot be performed, use item 4.c.

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

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

For a particular measurement system, which may include radiochemical separation :

$$LLD = \frac{4.66\ s_b}{\overline{E} \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

$\overline{E}$ is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y and Δt should be used in the calculation.

TABLE 4.12-2 (Continued)

TABLE NOTATION

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. Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 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.

APPLICABILITY: At all times.

ACTION:

a. 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 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 25 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, 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. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 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 Operating Report pursuant to Specification 6.9.1.8.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.12-1.4c shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials (which contain nuclides produced at nuclear power stations) supplied as part of an Interlaboratory Comparison Program, described in the ODCM, that has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be maintained as described in the ODCM.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (ATV) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Section 50.72 to 10 CFR Part 50 with a followup report pursuant to Section 50.73 to 10 CFR Part 50. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are generally consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 30 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDRY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

MEETING FREQUENCY

6.5.1.4 The SNSOC shall meet at least once per calendar month and as convened by the SNSOC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all procedures required by Specifications 6.8.1, 6.8.2 and 6.8.3 and changes thereto, 2) all programs required by Specification 6.8.4 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

f. Review of all REPORTABLE EVENTS and Special Reports.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Operations and to the Director-Safety Evaluation and Control.

l. Review and approve changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.1.7 The SNSOC shall:

a. Recommend to the Station Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control of disagreement between the SNSOC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

6.5.2 SAFETY EVALUATION AND CONTROL (SEC)

FUNCTION

6.5.2.1 SEC shall function to provide independent review of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Administrative controls and quality assurance practices

i. Other appropriate fields associated with the unique characteristics of the nuclear power plant

COMPOSITION

6.5.2.2 The SEC staff shall be composed of the Director-Safety Evaluation and Control and a minimum of three individuals who are qualified as staff specialists. Each SEC staff specialist shall have an academic degree in an engineering or physical science field and, in addition, shall have a minimum of five years technical experience in one or more areas given in Specification 6.5.2.1. These staff specialists shall not be directly involved in the licensing function.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the Director-Safety Evaluation and Control to provide expert advice to the SEC.

MEETING FREQUENCY

6.5.2.5 The SEC staff shall meet at least once per calendar month for the purpose of fostering interaction of reviews regarding safety-related operational activities.

REVIEW

6.5.2.7 The following subjects shall be reviewed by SEC:

a. Written safety evaluations of changes in the stations as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2) and is accomplished by review of minutes of the Station Nuclear Safety and Operating Committee and the design change program.

b. Proposed changes in procedures, proposed changes in the station, or proposed tests or experiments, any of which may involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Matters of this kind shall be referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee following its review prior to implementation.

c. Changes in the technical specifications or license amendments relating to nuclear safety prior to implementation except in those cases where the change is identical to a previously reviewed proposed change.

d. Violations, REPORTABLE EVENTS and Special Reports such as:

1. Violations of applicable codes, resulations, orders, Technical Specifications, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations from normal or expected performance of station safety-related structures, systems, or components; and

3. All REPORTABLE EVENTS submitted in accordance with Section 50.73 to 10 CFR Part 50 and Special Reports required by Specification 6.9.2.

Review of events covered under this paragraph shall include the results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. The Quality Assurance Department audit program at least once per 12 months and audit reports.

f. Any other matter involving safe operation of the nuclear power stations which is referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee.

g. Reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

AUTHORITY

6.5.2.9 The Director-Safety Evaluation and Control shall report to and advise the Manager-Nuclear Technical Services, who shall advise the Vice President-Nuclear Operations on those areas of responsibility specified in Section 6.5.2.7.

RECORDS

6.5.2.10 Records of SEC activities required by Section 6.5.2.7 shall be prepared and maintained in the SEC files and a summary shall be disseminated as indicated below each calendar month.

1. Vice President-Nuclear Operations
2. Nuclear Power Station Managers
3. Manager-Nuclear Operations and Maintenance
4. Manager-Nuclear Technical Services
5. Manager-Quality Assurance, Operations
6. Others that the Director-Safety Evaluation and Control may designate.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SNSOC and the results of this review shall be submitted to the Director-Safety Evaluation and Control and the Manager - Nuclear Operations and Maintenance.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the SNSOC and approved by the Station Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

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

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all control point chemistry conditions,

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action, and

(vii) Monitoring of the condensate at the discharge of the condensate pumps for evidence of condenser inleakage. When condenser in-leakage is confirmed, the leak shall be repaired, plugged, or isolated within 96 hours.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details requested in license conditions based on other commitments shall be included in this report.

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

ANNUAL REPORTS 1/

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

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

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

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CORE SURVEILLANCE REPORT

6.9.1.7 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) in all core planes containing Bank "D" control rods and in all unrodded core planes, the surveillance power level, P_m, for Technical Specifications 3.2.1 and 3.2.6, and the F_Q flyspeck basis as determined using the definitions and methodology in WCAP-8385 and Westinghouse letter to NRC dated April 6, 1978, Serial No. NS-CE-1749 shall be provided to the Regional Administrator, Region II, with a copy to:

Director, Office of Nuclear Reactor Regulation
Attention: Chief, Core Performance Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

at least 60 days prior to cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new submittal or an amended submittal to the Core Surveillance Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter.

Any additional information needed to support the F_{xy}^{RTP} and P_m submittal will be by request from the NRC and need not be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.8 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison (as appropriate) with preoperational studies, 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 land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include 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 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 individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1 and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

* A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.9 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report shall be submitted within 60 days after January 1 of each year. This report shall include an assessment of the radiation doses to the maximum exposed MEMBERS OF THE PUBLIC due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual Meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Concurrent meteorological conditions or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

If the dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specifications 3.11.1.2a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed MEMBER OF THE PUBLIC shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operations.

The Radioactive Effluent Release Reports shall include a list of unplanned releases as required to be reported in Section 50.73 to 10 CFR Part 50 from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirement of the applicable reference specification:

a. Inservice Inspection Program Reviews shall be reported within 90 days of completion. Specification 4.0.5.

b. ECCS Actuation shall be reported within 90 days of the occurrence. The report shall describe the circumstances of the actuation and the total accumulated cycles to date. Specifications 3.5.2 and 3.5.3.

c. With the primary coolant specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100 $\bar{E}$ Ci/gram, a specific activity analysis shall be included in the Special Report. The information requested in Specification 3.4.8 shall also be included in that report. |

d. With sealed sources or fission detector leakage tests revealing the presence of $\geq$0.005 microcuries of removable contamination submit a special report on an annual basis outlining the corrective actions taken to prevent the spread of contamination. Specification 4.7.11.1.3.

e. With the MTC more positive than 0 $\Delta k/k/^{o}F$ submit a special report within the next 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. Specification 3.1.1.4.

f. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10.1, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.1.

g. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6.2, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6.2. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.2.

h. Inoperable Fire Detection Instrumentation, Specification 3.3.3.7.

i. Inoperable fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2, 3.7.14.3, 3.7.14.4 and 3.7.14.5.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS and Special Reports.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material release to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.7-1.