November 13, 1990

Docket Nos. 50-352
and 50-353

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Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
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Dear Mr. Hunger:

SUBJECT: RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (TSCR 89-07)
LIMERICK GENERATING STATION, UNITS 1 AND 2
(TAC NOS. 76908 AND 76909)

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. NPF-39 and Amendment No. 11 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 29, 1990, as supplemented by letter dated October 19, 1990.

These amendments revise the TSs in accordance with the guidance specified in NRC Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls of Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program."

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 48 to
   License No. NPF-39
   Amendment No. 11 to
   License No. NPF-85
2. Safety Evaluation

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See next page

DOCUMENT NAME: TAC NOS. 76908 AND 76909

PDI-2/LA	PRBP	DRPB	PDI-2/PM	OGC	PDI-2/D	C/PRBP
MO'Brien	JLevine	TEssig	RClark	L. DEWEY	WButler	LCunningham
11/8/90	10/17/90	10/19/90	10/19/90	11/5/90	11/8/90	10/19/90

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

November 13, 1990

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and 50-353

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
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Richard J. Clark

Richard J. Clark, Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

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Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated May 29, 1990, as supplemented by letter dated October 19, 1990, 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-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 48 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective January 2, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1990

*See previous concurrence

PDI-2/LA	PDI-2/PM*	OGC	PDI-2/D
MO'Brien	RClark	LDEWEY	WButler
11/8/90	08/10/90 10/19/90	11/5/90	11/8/90

3. This license amendment is effective January 2, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf page(s) are provided to maintain document completeness.*

Remove	Insert
i	i*
ii	ii
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x	x
xv	xv*
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xviii	xviii*
xix	xix
xx	xx*
xxi	xxi*
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This total system response time consists of two components, the instrumentation response time and the breaker arc suppression time. These times may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.13 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.14 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.17 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.18 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.19 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.20 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.21 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.22 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.23 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel).

OFFSITE DOSE CALCULATION MANUAL

1.24 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.25 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the solidification or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With solidification or dewatering, the PCP shall identify the process parameters influencing solidification or dewatering, based on laboratory scale and full scale testing or experience.

PURGE - PURGING

1.31 PURGE or PURGING shall be 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 required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.1-1 of Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.35 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.2-1 of Specification 3.6.5.2.2.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

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

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.39 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

1.40 (Deleted)

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.42 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

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

UNIDENTIFIED LEAKAGE

1.44 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.45 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which 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

1.46 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any 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

1.47 VENTING shall be 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.

Section 3.3.7.3 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-74 THROUGH 3/4 3-75 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown system instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

4.3.7.4.2 Each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, 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 channel(s) to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

Section 3.3.7.11 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-99 THROUGH 3/4 3-102 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.

b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas 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 4.3.7.12-1.

TABLE 3.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	110
2. (Deleted)			
3. (Deleted)			
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR			
a. Noble Gas Activity Monitor	1	**	115
5. (Deleted)			

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATIONS

* (Deleted)

** During operation of the main condenser steam jet air ejector and offgas treatment system.

*** (Deleted)

ACTION STATEMENTS

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 111-114 (Deleted)

ACTION 115 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases to the environment may continue for up to 72 hours provided that the North Stack Effluent Noble Gas Activity Monitor is OPERABLE; otherwise, be in at least HOT SHUTDOWN within 12 hours.

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TABLE 4.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
2. (Deleted)					
3. (Deleted)					
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (STEAM JET AIR EJECTOR)					
a. Noble gas activity monitor	D	M	R(2)	Q(1)	**
5. (Deleted)					

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATIONS

* (Deleted)

** During operation of the main condenser steam jet air ejector and offgas treatment system.

*** (Deleted)

(1) 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. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), previously National Bureau of Standards, or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. 0.0 volume percent hydrogen, balance nitrogen, and
2. 4 volume percent hydrogen, balance nitrogen.

(4) (Deleted)

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3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.8 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one turbine control valve and/or one turbine stop valve per high pressure turbine steam lead inoperable and/or with one turbine combined intermediate valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam lead(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

4.3.8.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each of the following valves through at least one complete cycle from the running position:

a) For the overspeed protection control system;

1) Six low pressure turbine intercept valves

b) For the electrical overspeed trip system and the mechanical overspeed trip system;

1) Four high pressure turbine stop valves, and

2) Six low pressure turbine intermediate stop valves.

Section 3/4 11.1.1 through 3/4 11.1.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-2 THROUGH 3/4 11-6 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above 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 and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

Section 3/4 11.2.1 through Section 3/4 11.2.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-9 THROUGH 3/4 11-14 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

Section 3/4 11-2.7 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM.

Section 3/4 11.3 and Section 3/4 11.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE PCP OR ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-19 THROUGH 3/4 11-20 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

Section 3/4.12 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 12-2 THROUGH 3/4 12-14 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression, and the response time of the system logic.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to be OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output in the resultant heat balance calculation (P-1) with date obtained during a previous heat balance calculation (P-1).

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.11 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.12 OFFGAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures and noble gases in the off-gas system.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

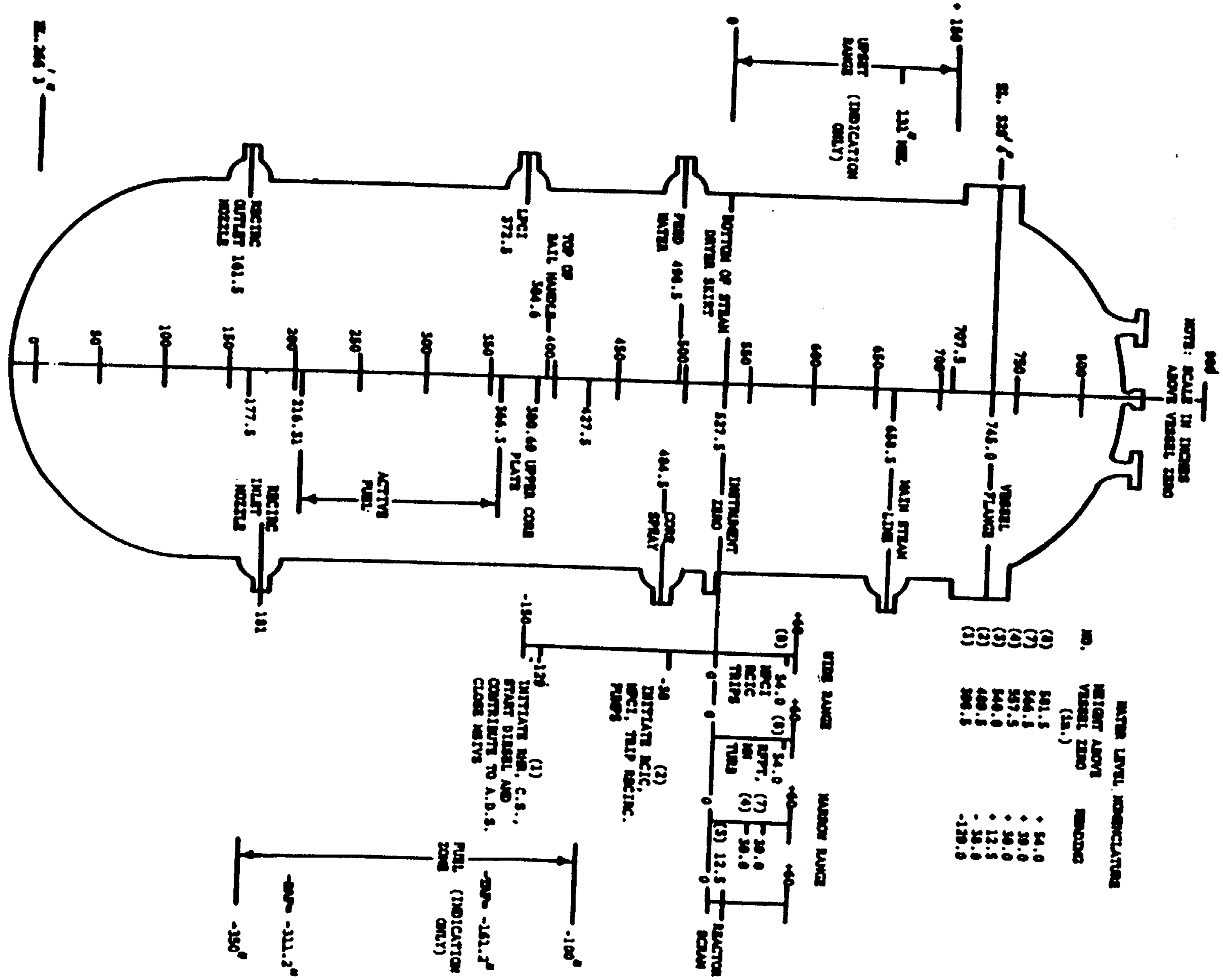
This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.

Wide Range Level

This indication is reactor coolant temperature sensitive. The calibration is thus made at rated conditions. The level error at low pressures (temperatures) is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not indicated level.



BASES FIGURE B 3/4.3-1

REACTOR VESSEL WATER LEVEL

3/4.11.1.1 and 3/4.11.1.2 (Deleted)

THE INFORMATION FROM THESE SECTIONS
HAS BEEN RELOCATED TO THE ODCM.

3/4.11.1.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste 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 treatment system.

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

3/4.11.2.1 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.11.2.2 through 3/4.11.2.4 (Deleted)

THE INFORMATION FROM THESE SECTIONS
HAS BEEN RELOCATED TO THE ODCM.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas treatment system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 MAIN CONDENSER

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

3/4 11.2.7, 3/4 11.3, and 3/4 11.4 (Deleted) - INFORMATION FROM THESE SECTIONS RELOCATED TO THE ODCM OR PCP.

Section 3/4.12 (Deleted)

THE INFORMATION FROM THIS SECTION HAS BEEN RELOCATED TO THE ODCM. BASES PAGE B 3/4 12-2 HAS BEEN INTENTIONALLY OMITTED.

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBER OF THE PUBLIC, shall be as shown in Figures 5.1.3-1a and 5.1.3-1b.

5.1.4 (Deleted)

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined reinforced concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in a shape of a truncated cone on top of a water filled suppression chamber and is separated by a diaphragm slab and connected to the suppression chamber through a series of downcomer vents. The drywell has a maximum free air volume of 243,580 cubic feet at a minimum suppression pool level of 22 feet. The suppression chamber has a maximum air region of 159,540 cubic feet and a minimum water region of 122,120 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 55 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 220°F.

c. Maximum external to internal differential pressure 5 psid.

d. Maximum floor differential pressure: 30 psid, downward.
20 psid, upward.

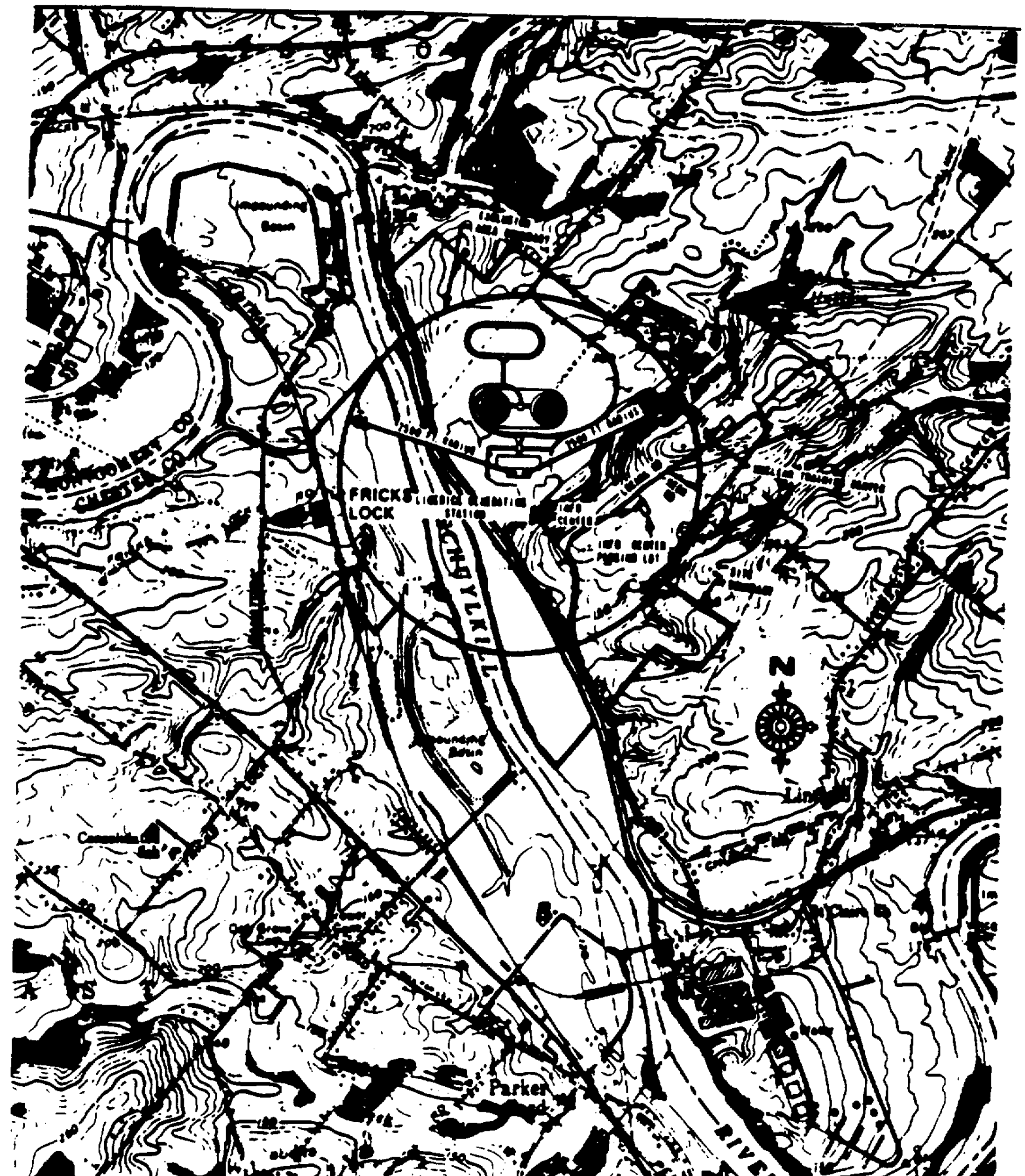


FIGURE 5.1.1-1

EXCLUSION AREA

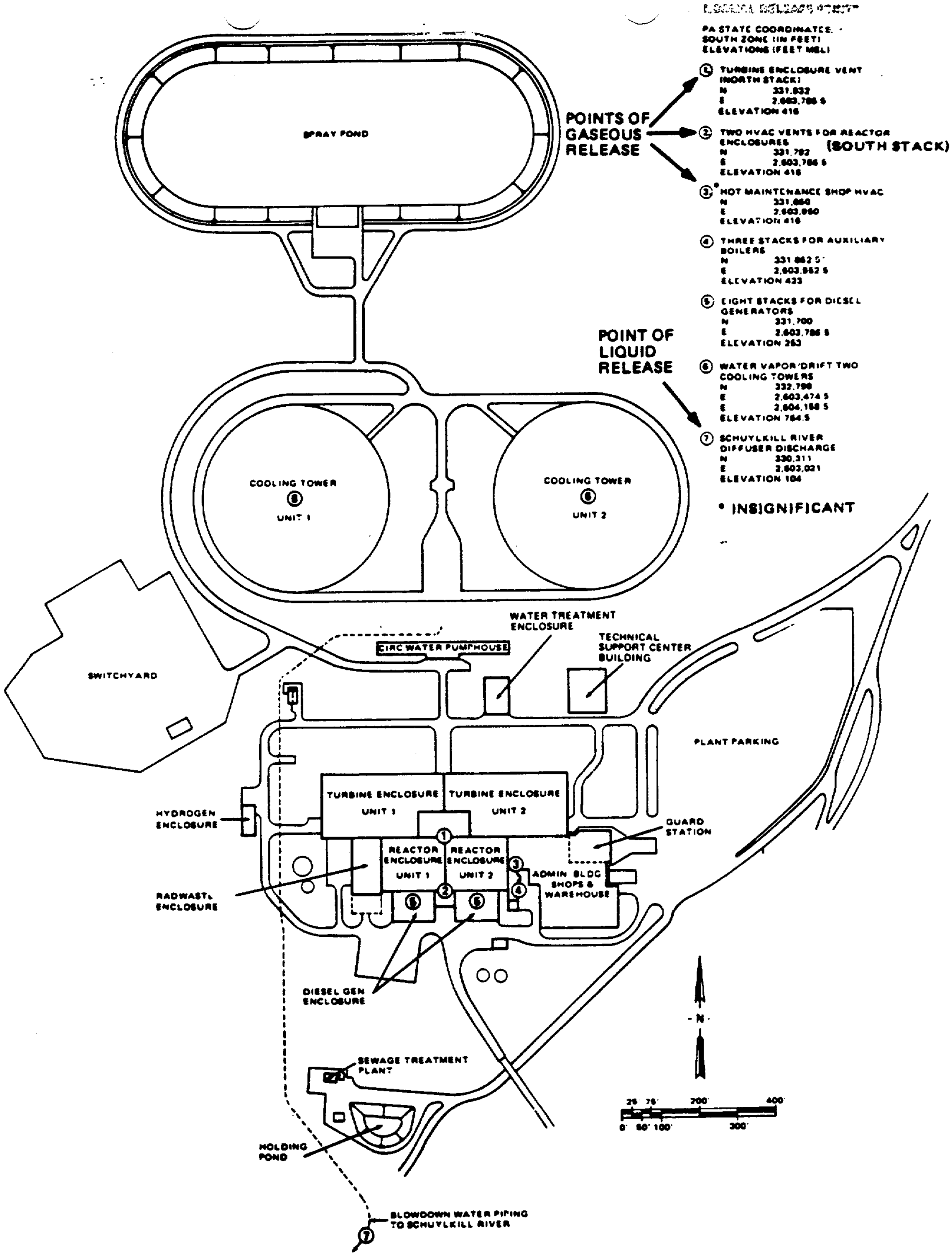


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

THE FIGURE ON THIS PAGE HAS BEEN RELOCATED TO THE ODCM.

PROCEDURES AND PROGRAMS (Continued)

d. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

PROCEDURES AND PROGRAMS (Continued)

8) Limitations on the annual quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable, and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

e. Meteorological Monitoring Program

A program shall be provided to provide meteorological information in the environs of the plant. The program shall provide sufficient meteorological data for estimating potential radiation doses to the public.

The program shall (1) be contained in the ODCM, (2) conform to the guidance of Regulatory Guide 1.23, "Safety Guide 23 - Onsite Meteorlogical Program", and (3) include limitations on the operability of meteorological monitoring instrumentation including surveillance tests in accordance with the methodology in the ODCM.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

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 Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendme : 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 unit.

6.9.1.2 The startup report shall address each of the tests identified in Subsection 14.2.12 of the Final Safety Analysis Report 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 required 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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

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** (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

*A single submittal may be made for a multiple unit station.
**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL REPORTS (Continued)

dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves; and

c. Any other unit unique reports required on an annual basis.

d. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS*

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the the main steam system safety/relief valves, shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 The Semiannual Radioactive Effluent Release Report 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 report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

*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.

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6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit 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.

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

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

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

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit 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 radiation exposure for all individuals entering radiation control areas.

RECORD RETENTION (Continued)

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.6.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual not listed in Section 6.10.2.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the NRB.

l. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/h but less than 1000 mrem/h shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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

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

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrems shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of Shift Supervision on duty and/or the health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3n. This documentation shall contain:

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

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (Continued)

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3n. This documentation shall contain:

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

2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

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

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

6.15 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 11
License No. NPF-85

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated May 29, 1990, as supplemented by letter dated October 19, 1990, 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-85 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 11 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective January 2, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1990

*See previous concurrence

PDI-2/LA	PDI-2/PM*	* OGC	PDI-2/D
MO'Brien	RClark	L. Dewey	WButler
11/8/90	08/10/90 10/19/90	11/15/90	11/8/90

3. This license amendment is effective January 2, 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects - I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 13, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf page(s) are provided to maintain document completeness.*

Remove	Insert
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xviii	xviii*
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This total system response time consists of two components, the instrumentation response time and the breaker arc suppression time. These times may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.13 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.14 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.17 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.18 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.19 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.20 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.21 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.22 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.23 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core (for each class of fuel).

OFFSITE DOSE CALCULATION MANUAL

1.24 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.25 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.26 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.27 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.28 PRESSURE BOUNDARY LEAKAGE shall be leakage through a nonisolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.29 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. The primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the provisions to assure that the SOLIDIFICATION or dewatering and packaging of radioactive wastes results in a waste package with properties that meet the minimum and stability requirements of 10 CFR Part 61 and other requirements for transportation to the disposal site and receipt at the disposal site. With SOLIDIFICATION or dewatering, the PCP shall identify the process parameters influencing SOLIDIFICATION or dewatering based on laboratory scale and full scale testing or experience.

PURGE - PURGING

1.31 PURGE or PURGING shall be 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 required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3293 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.1-1 of Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.35 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.5.2.2-1 of Specification 3.6.5.2.2.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY (Continued)

b. All refueling floor secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. At least one door in each access to the refueling floor secondary containment is closed.

e. The sealing mechanism associated with each refueling floor secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

f. The pressure within the refueling floor secondary containment is less than or equal to the value required by Specification 4.6.5.1.2a.

REPORTABLE EVENT

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

ROD DENSITY

1.37 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SHUTDOWN MARGIN

1.38 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.39 The SITE BOUNDARY shall be that line as defined in Figure 5.1.3-1a.

1.40 (Deleted)

SOURCE CHECK

1.41 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.42 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

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

UNIDENTIFIED LEAKAGE

1.44 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

UNRESTRICTED AREA

1.45 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which 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

1.46 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any 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

1.47 VENTING shall be 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.

Section 3.3.7.3 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-74 THROUGH 3/4 3-75 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

LIMITING CONDITION FOR OPERATION

3.3.7.4 The remote shutdown system instrumentation and controls shown in Table 3.3.7.4-1 shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the number of OPERABLE remote shutdown system instrumentation channels less than required by Table 3.3.7.4-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown system controls less than required in Table 3.3.7.4-1, restore the inoperable control(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.4.1 Each of the above required remote shutdown monitoring instrumentation channels shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.7.4-1.

4.3.7.4.2 Each of the above remote shutdown control switch(es) and control circuits shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one or more loose-part detection system channels inoperable for more than 30 days, 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 channel(s) to OPERABLE status.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

Section 3.3.7.11 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 3-99 THROUGH 3/4 3-102 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

INSTRUMENTATION

OFFGAS GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.12 The offgas monitoring instrumentation channels shown in Table 3.3.7.12-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specifications 3.11.2.5 and 3.11.2.6 respectively, are not exceeded.

APPLICABILITY: As shown in Table 3.3.7.12-1

ACTION:

a. With an offgas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable, and take the ACTION shown in Table 3.3.7.12-1.

b. With less than the minimum number of offgas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.12-1. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain why this inoperability was not corrected in a timely manner in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.3.7.12 Each offgas 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 4.3.7.12-1.

TABLE 3.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	110
2. (Deleted)			
3. (Deleted)			
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR			
a. Noble Gas Activity Monitor	1	**	115
5. (Deleted)			

TABLE 3.3.7.12-1 (Continued)

TABLE NOTATIONS

* (Deleted)

** During operation of the main condenser steam jet air ejector and offgas treatment system.

*** (Deleted)

ACTION STATEMENTS

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

ACTION 111-114 (Deleted)

ACTION 115 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, releases to the environment may continue for up to 72 hours provided that the North Stack Effluent Noble Gas Activity Monitor is OPERABLE; otherwise, be in at least HOT SHUTDOWN within 12 hours.

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TABLE 4.3.7.12-1

OFFGAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
2. (Deleted)					
3. (Deleted)					
4. MAIN CONDENSER OFFGAS PRE-TREATMENT RADIOACTIVITY MONITOR (STEAM JET AIR EJECTOR)					
a. Noble gas activity monitor	D	M	R(2)	Q(1)	**
5. (Deleted)					

TABLE 4.3.7.12-1 (Continued)

TABLE NOTATIONS

* (Deleted)

** During operation of the main condenser steam jet air ejector and offgas treatment system.

*** (Deleted)

(1) 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. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Institute of Standards and Technology (NIST), previously National Bureau of Standards, or using standards that have been obtained from suppliers that participate in measurement assurance activities with NIST. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. 0.0 volume percent hydrogen, balance nitrogen, and
2. 4 volume percent hydrogen, balance nitrogen.

(4) (Deleted)

INTENTIONALLY LEFT BLANK

INSTRUMENTATION

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.8 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one turbine control valve and/or one turbine stop valve per high pressure turbine steam lead inoperable and/or with one turbine combined intermediate valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours or close at least one valve in the affected steam lead(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.8.1 The provisions of Specification 4.0.4 are not applicable.

4.3.8.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each of the following valves through at least one complete cycle from the running position:

a) For the overspeed protection control system;

1) Six low pressure turbine intercept valves

b) For the electrical overspeed trip system and the mechanical overspeed trip system;

1) Four high pressure turbine stop valves, and

2) Six low pressure turbine intermediate stop valves.

Section 3/4 11.1.1 through 3/4 11.1.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-2 THROUGH 3/4 11-6 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above 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 and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

Section 3/4 11.2.1 through Section 3/4 11.2.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-9 THROUGH 3/4 11-14 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

Section 3/4 11-2.7 (Deleted)

THE INFORMATION FROM THIS
TECHNICAL SPECIFICATIONS
SECTION HAS BEEN RELOCATED
TO THE ODCM.

Section 3/4 11.3 through 3/4 11.4 (Deleted)

THE INFORMATION FROM THESE TECHNICAL SPECIFICATIONS SECTIONS HAS BEEN RELOCATED TO THE PCP OR ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 11-19 THROUGH 3/4 11-20 OF THESE SECTIONS HAVE BEEN INTENTIONALLY OMITTED.

Section 3/4.12 (Deleted)

THE INFORMATION FROM THIS TECHNICAL SPECIFICATIONS SECTION HAS BEEN RELOCATED TO THE ODCM. TECHNICAL SPECIFICATIONS PAGES 3/4 12-2 THROUGH 3/4 12-14 OF THIS SECTION HAVE BEEN INTENTIONALLY OMITTED.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, NEDO-24222, dated December 1979, and Section 15.8 of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a supplement to the reactor trip. During turbine trip and generator load rejection events, the EOC-RPT will reduce the likelihood of reactor vessel level decreasing to level 2. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 175 ms. Included in this time are: the response time of the sensor, the time allotted for breaker arc suppression, and the response time of the system logic.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel. This instrumentation does not provide actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits and Section 3/4.3 Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 (Deleted) INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50, Appendix A. The Unit 1 RHR transfer switches are included only due to their potential impact on the RHRSW system, which is common to both units.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to be OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output in the resultant heat balance calculation (P-1) with data obtained during a previous heat balance calculation (P-1).

3/4.3.7.8 CHLORINE AND TOXIC GAS DETECTION SYSTEMS

The OPERABILITY of the chlorine and toxic gas detection systems ensures that an accidental chlorine and/or toxic gas release will be detected promptly and the necessary protective actions will be automatically initiated for chlorine and manually initiated for toxic gas to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically be placed in the chlorine isolation mode of operation to provide the required protection. Upon detection of a high concentration of toxic gas, the control room emergency ventilation system will manually be placed in the chlorine isolation mode of operation to provide the required protection. The detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95 "Protection of Nuclear Power Plant Control Room Operators against an Accidental Chlorine Release," February 1975.

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.11 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.3.7.12 OFFGAS MONITORING INSTRUMENTATION

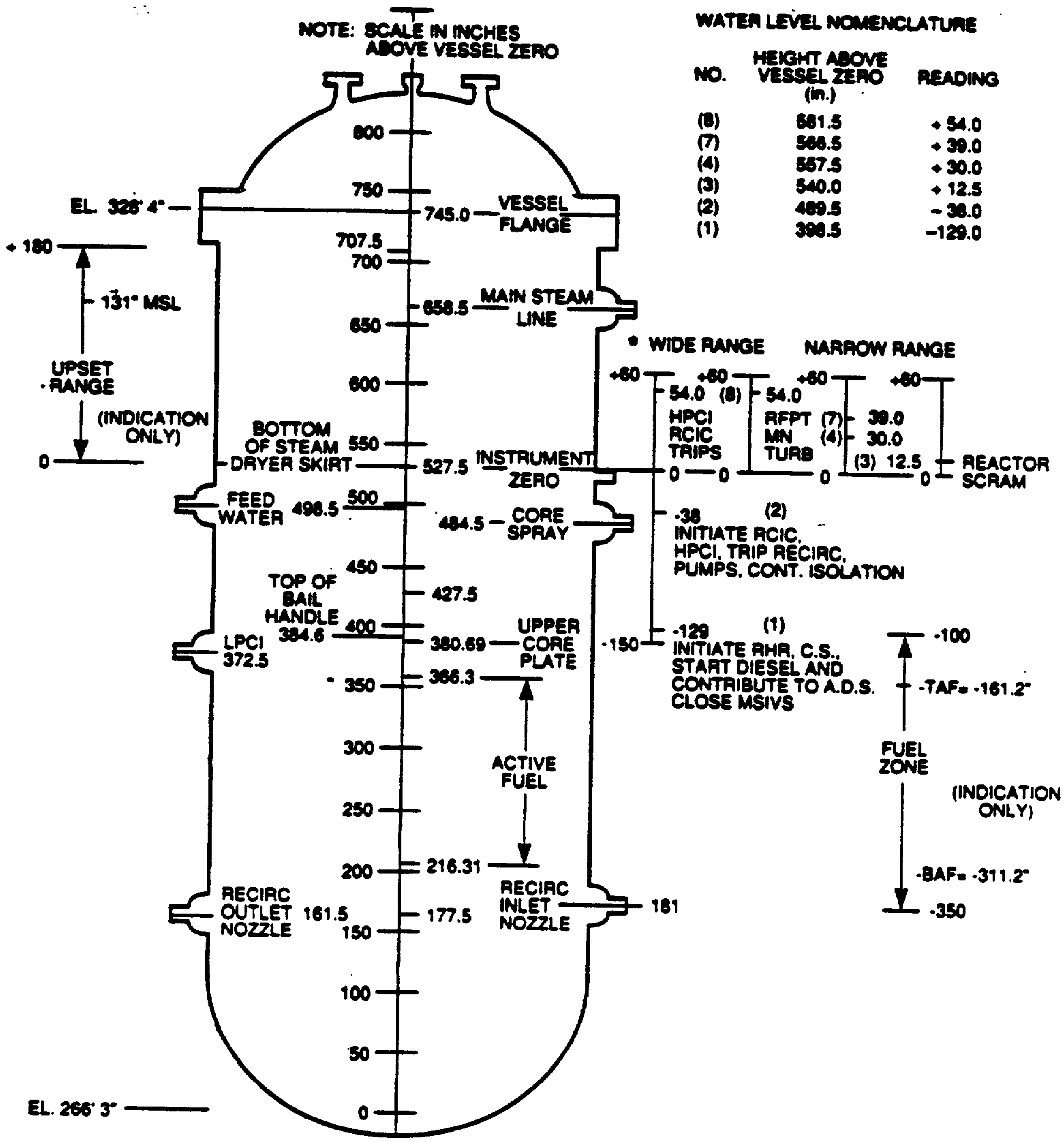
This instrumentation includes provisions for monitoring the concentrations of potentially explosive gas mixtures and noble gases in the off-gas system.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate action of the feedwater system/main turbine trip system in the event of failure of feedwater controller under maximum demand.



WATER LEVEL NOMENCLATURE

NO.	HEIGHT ABOVE VESSEL ZERO (in.)	READING
(8)	581.5	+ 54.0
(7)	566.5	+ 39.0
(4)	557.5	+ 30.0
(3)	540.0	+ 12.5
(2)	489.5	− 38.0
(1)	398.5	−129.0

* Wide Range Level

This indication is reactor coolant temperature sensitive. The calibration is thus made at rated conditions. The level error at low pressures (temperatures) is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not indicated level.

BASES FIGURE B 3/4.3-1

REACTOR VESSEL WATER LEVEL

3/4.11.1.1 and 3/4.11.1.2 (Deleted)

THE INFORMATION FROM THESE SECTIONS
HAS BEEN RELOCATED TO THE ODCM.

3/4.11.1.3 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste 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 treatment system.

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

3/4.11.2.1 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

3/4 11.2.2 through 3/4 11.2.4 (Deleted)

THE INFORMATION FROM THESE SECTIONS
HAS BEEN RELOCATED TO THE ODCM.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the main condenser offgas treatment system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen below its flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 MAIN CONDENSER

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

3/4.11.2.7, 3/4.11.3, and 3/4.11.4 (Deleted) - INFORMATION FROM THESE SECTIONS RELOCATED TO THE ODCM OR PCP.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

Section 3/4.12 (Deleted)

THE INFORMATION FROM THIS SECTION
HAS BEEN RELOCATED TO THE ODCM.
BASES PAGE B 3/4 12-2 HAS BEEN
INTENTIONALLY OMITTED.

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

MAPS DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBER OF THE PUBLIC, shall be as shown in Figures 5.1.3-1a and 5.1.3-1b.

5.1.4 (Deleted)

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined reinforced concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in a shape of a truncated cone on top of a water filled suppression chamber and is separated by a diaphragm slab and connected to the suppression chamber through a series of downcomer vents. The drywell has a maximum free air volume of 243,580 cubic feet at a minimum suppression pool level of 22 feet. The suppression chamber has a maximum air region of 159,540 cubic feet and a minimum water region of 122,120 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 55 psig.

b. Maximum internal temperature: drywell 340°F.
suppression pool 220°F.

c. Maximum external to internal differential pressure 5 psid.

d. Maximum floor differential pressure: 30 psid, downward.
20 psid, upward.

FIGURE 5.1.1-1

EXCLUSION AREA

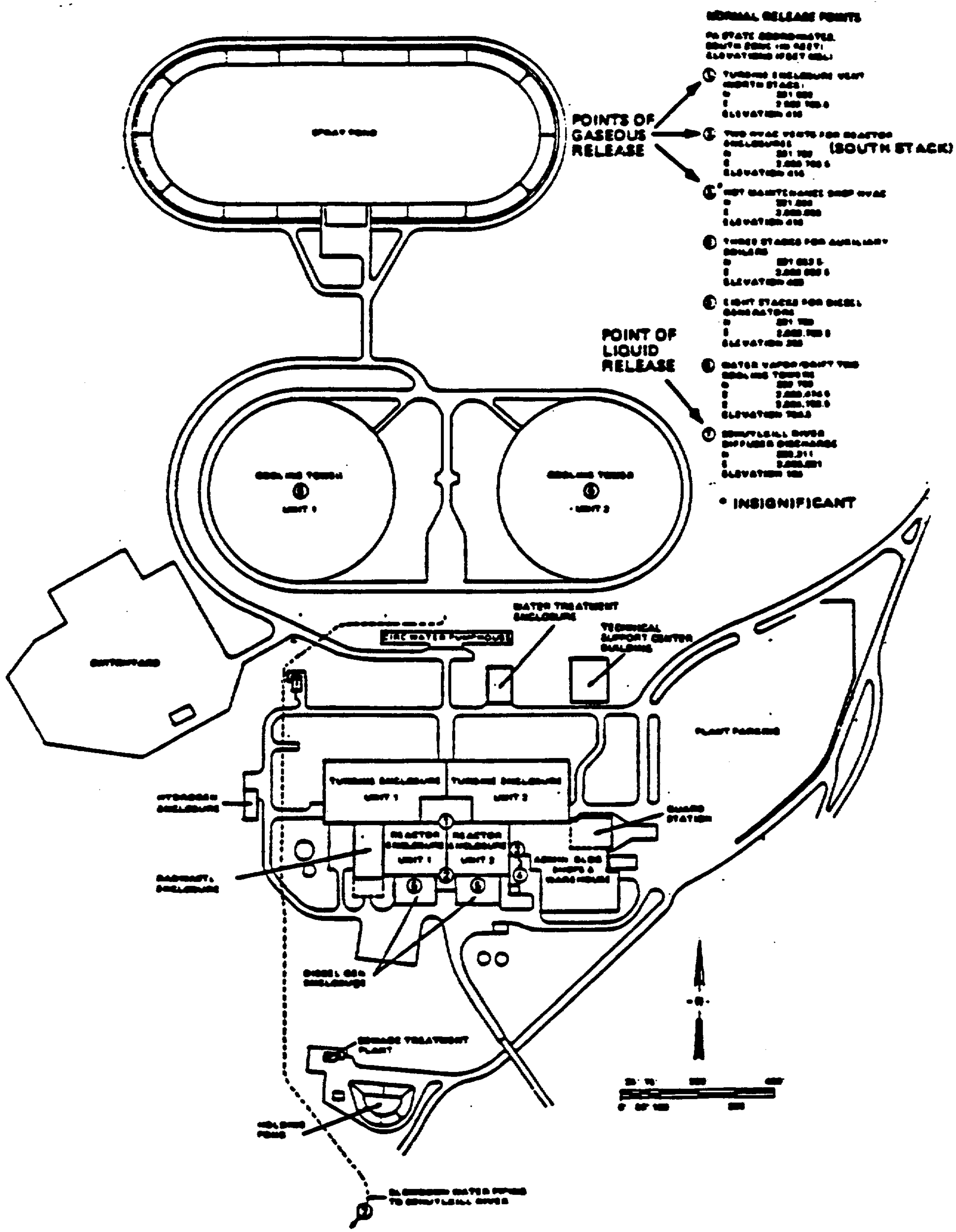


FIGURE 5.1.3-1b

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY
FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

THE FIGURE ON THIS PAGE HAS BEEN RELOCATED TO THE ODCM.

PROCEDURES AND PROGRAMS (Continued)

d. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR Part 20, Appendix B, Table II, Column 1,

PROCEDURES AND PROGRAMS (Continued)

8) Limitations on the annual quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on venting and purging of the Mark II containment through the Standby Gas Treatment System to maintain releases as low as reasonably achievable, and

11) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

e. Meteorological Monitoring Program

A program shall be provided to provide meteorological information in the environs of the plant. The program shall provide sufficient meteorological data for estimating potential radiation doses to the public.

The program shall (1) be contained in the ODCM, (2) conform to the guidance of Regulatory Guide 1.23, "Safety Guide 23 - Onsite Meteorological Program", and (3) include limitations on the operability of meteorological monitoring instrumentation including surveillance tests in accordance with the methodology in the ODCM.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

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 Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in Subsection 14.2.12 of the Final Safety Analysis Report 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 required 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 operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

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** (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

*A single submittal may be made for a multiple unit station.
**This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL REPORTS (Continued)

dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. Documentation of all challenges to safety/relief valves; and

c. Any other unit unique reports required on an annual basis.

d. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS*

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the the main steam system safety/relief valves, shall be submitted on a monthly basis to the ATTN: Document Control Desk, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 The Semiannual Radioactive Effluent Release Report 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 initial report shall be submitted during first report period, as described above, following initial criticality. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

*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.

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6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit 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.

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

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

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

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit 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 radiation exposure for all individuals entering radiation control areas.

RECORD RETENTION (Continued)

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.6.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Operational Quality Assurance Manual not listed in Section 6.10.2.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PORC and the NRB.

l. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analysis required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/h but less than 1000 mrem/h shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

*Health physics personnel or personnel escorted by health physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

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

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

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrems shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of Shift Supervision on duty and/or the health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, continuous surveillance direct or remote (such as use of closed circuit TV cameras), may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3n. This documentation shall contain:

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

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (Continued)

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.3n. This documentation shall contain:

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

2. A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

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

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

6.15 (Deleted) - INFORMATION FROM THIS SECTION RELOCATED TO THE ODCM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 48 AND 11 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated May 29, 1990, as supplemented by letter dated October 19, 1990, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License Nos. NPF-39 and NPF-85 for the Limerick Generating Station, Units 1 and 2. These proposed amendments would revise the Technical Specifications (TS) in accordance with the guidance specified in NRC Generic Letter 89-01, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls of Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program." Generic Letter (GL) 89-01 suggests that licensees 1) implement programmatic controls for Radiological Effluent Technical Specifications (RETS) in the Administrative Controls section of TS, and 2) relocate procedural details of RETS to the Offsite Dose Calculation Manual (ODCM) or to the Process Control Program (PCP). In addition, the licensee is proposing that TS section 3/4 3.7.3, Meteorological Monitoring Instrumentation" be relocated to the ODCM.

In the application of May 29, 1990, the licensee had also proposed relocating two site maps from Section 5.1.3.1a and 5.1.3.1b to the ODCM. The letter of October 19, 1990 withdrew this proposed change and does not affect the no significant hazards consideration.

As part of the proposed changes, Table 4.11.2.1.2-1, "Radioactive Gaseous Waste Sampling and Analysis Program" (page 3/4 11-9) is being moved to the ODCM. This table lists the potential gaseous release points, the sampling frequency (e.g., grab samples or continuous in-line monitors), the minimum analysis frequency, the type of activity analysis and the lower limit of detection. In the application of May 29, 1990, PECo had proposed deleting the condenser Off Gas Pretreatment Monitor from this table prior to relocating the table to the ODCM, since this instrument is not monitoring a potential point of release to the environment. PECo's supplemental letter of October 19, 1990 withdrew the request to remove this monitor from Table 4.11.2.1.2-1 so the table as it now exists is being moved to the ODCM. The licensee may at some future date remove this monitor from the table in accordance with the provisions in Section 6.14.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided in Generic Letter 89-01 and are addressed below.

(1) The licensee has proposed to incorporate programmatic controls for radioactive effluents and radiological environmental monitoring in Specification 6.8.4, "Procedures and Programs," of the TS as noted in the guidance provided in Generic Letter 89-01. The programmatic controls ensure that programs are established, implemented, and maintained to ensure that operating procedures are provided to control radioactive effluents consistent with the requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50.

(2) The licensee has confirmed that the detailed procedural requirements addressing Limiting Conditions for Operation, their applicability, remedial actions, associated surveillance requirements, or reporting requirements have been prepared to implement the relocation of these procedural details to the ODCM or PCP. These changes to the ODCM and PCP have been prepared in accordance with the new Administrative Controls in the TS on changes to the ODCM and PCP so that they will be implemented in the ODCM or PCP when this amendment is effective.

These procedural details that have been removed from the TS are not required by the Commission's regulations to be included in TS. They have been prepared for incorporation in the ODCM or PCP upon the effective date of this license amendment and may be subsequently changed by the licensee without prior NRC approval. Changes to the ODCM and PCP are documented and will be retained for the duration of the operating license in accordance with Specification 6.10.3n.

(3) The licensee has proposed replacing the existing specifications in the Administrative Controls section of the TS for the Annual Radiological Environmental Operating Report, Specification 6.9.1.7, for the Semiannual Radioactive Effluent Release Report, Specification 6.9.1.8, for the Process Control Program, Specification 6.13, and for the Offsite Dose Calculation Manual, Specification 6.14, with the updated specifications that were provided in Generic Letter 89-01.

The following specifications that are included under the heading of Radioactive Effluents have been retained in the TS. This is in accordance with the guidance of Generic Letter 89-01.

SPECIFICATION	TITLE
3/4.3.4.7.12	EXPLOSIVE GAS MONITORING INSTRUMENTATION (Retained existing requirements of this specification)
3/4.3.11.1.4	LIQUID HOLDUP TANKS
3/4.3.11.2.5	EXPLOSIVE GAS MIXTURE
3/4.3.11.2.6	MAIN CONDENSER

The relocation of the meterological monitoring program to the ODCM, while not suggested in GL 89-01, is in keeping with the intent of the generic letter. This change has been approved for other licensees and is acceptable.

On the basis of the above, the staff finds that the changes included in the proposed TS amendment request are consistent with the guidance provided in Generic Letter 89-01. Because the control of radioactive effluents continues to be limited in accordance with operating procedures that must satisfy the regulatory requirements of 10 CFR 20.106, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

The NRC staff with the knowledge and consent of the licensee made administrative corrections to the licensee's Technical Specification pages.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The staff has determined that these amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that these amendments involve no significant hazards consideration which was published in the FEDERAL REGISTER (55 FR 26291) on June 27, 1990 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations and (3) the issuance of these amendments will not be inimical to the common defense and the security nor to the health and safety of the public.

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Dated: November 13, 1990