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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 85
License No. NPF-11

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated May 22, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 85 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

RJ Barrett

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 1, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 85

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change. Pages indicated with an asterisk are provided for convenience.

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DEFINITIONS

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LINEAR HEAT GENERATION RATE

1.21 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

1.22 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

1.23 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

1.24 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 licensee, 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

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.26 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 Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.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 Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.27 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, normal and an emergency electrical power source, 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

1.28 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

1.29 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

1.30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.31 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. Each primary containment air lock is OPERABLE pursuant to 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 OPERABLE pursuant to 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

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.33 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

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 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.

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.

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All 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, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

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

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

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

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.39 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

1.40 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

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.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.44 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

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

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.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

TABLE 1.2

OPERATIONAL CONDITIONS

CONDITION	MODE SWITCH POSITION	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown# ***	> 200°F
4. COLD SHUTDOWN	Shutdown# ## ***	≤ 200°F
5. REFUELING*	Shutdown or Refuel** #	≤ 140°F

#The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

##The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.9.10.1.

*Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**See Special Test Exception 3.10.3

***The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided that the one-rod-out interlock is OPERABLE.

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3.3.7.10 INTENTIONALLY LEFT BLANK.

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.11 The explosive gas monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their Alarm/Trip setpoints set to ensure that the limits of specification 3.11.2.6 are not exceeded.

APPLICABILITY: During operation of the main condenser air ejector.

ACTION:

a. With an explosive gas 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 3.3.7.11-1.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation channels to an OPERABLE status wihtin 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.11 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies shown in Table 4.3.7.11-1.

TABLE 3.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)		
a. Hydrogen Monitor	1/train	110

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the 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. If the recombiner(s) temperature remains constant and THERMAL POWER has not changed, the grab sample collection frequency may be changed to 8 hours.

TABLE 4.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION*	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	D	M	Q	**

TABLE NOTATION

* The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

** During operation of the main condenser air ejector.

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.12 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 6.6.c 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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.

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.8.1-1.

4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.*

*The specified 18 month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1.

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level-High, Level 8	3

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	< 55.5 inches*	< 56.0 inches*

*See Bases Figure B 3/4 3-1.

TABLE 4.3.8.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-High, Level 8	S	M	R

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits as required by Table 3.3.7.11-1 of Specification 3.3.7.11.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.2 The release rate of the sum of the activities from the noble gases measured prior to the holdup line shall be limited to less than or equal to 3.4×10^5 microcuries/second.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the release rate of the sum of the activities of the noble gases prior to the holdup line exceeding 3.4×10^5 microcuries/second restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases prior to the holdup line shall be continuously monitored in accordance with Specification 3.3.7.11.

4.11.2.7.2 The release rate of the sum of the activities from noble gases prior to the holdup line shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken prior to the holdup line.

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the off gas pre-treatment Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

MONITORING INSTRUMENTATION (Continued)

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-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

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 should 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 (TIP) 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 specification allows use of substituted TIP data from symmetric channels if the control rod pattern is symmetric since the TIP data is adjusted by the plant computer to remove machine dependent and power level dependent bias. The source of data for the substitution may also be a 3-dimensional BWR core simulator calculated data set which is normalized to available real data. Since uncertainty could be introduced by the simulation and normalization process, an evaluation of the specific control rod pattern and core operating state must be performed to ensure that adequate margin to core operating limits is maintained.

3/4.3.7.8 DELETED

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 DELETED

3/4.3.7.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation provides for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

3/4.3.7.12 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."

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure, to prevent overfilling the reactor vessel which may result in high pressure liquid discharge through the safety/relief valve discharge lines.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 LIQUID HOLDUP TANKS

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 GASEOUS EFFLUENTS

3/4.11.2.1 EXPLOSIVE GAS MIXTURE

The specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits 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.

3/4.11.2.2 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.

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.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined post-stressed concrete vessel in the shape of a truncated cone closed by a steel dome. The drywell is above a cylindrical steel-lined post-stressed concrete suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 229,538 cubic feet. The suppression chamber has an air region of 164,800 to 168,100 cubic feet and a water region of 128,800 to 131,900 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

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

a. Maximum internal pressure 45 psig.

b. Maximum internal temperature: drywell 340°F.
suppression chamber 275°F.

c. Maximum external pressure 5 psig.

d. Maximum floor differential pressure: 25 psid, downward.
5 psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.

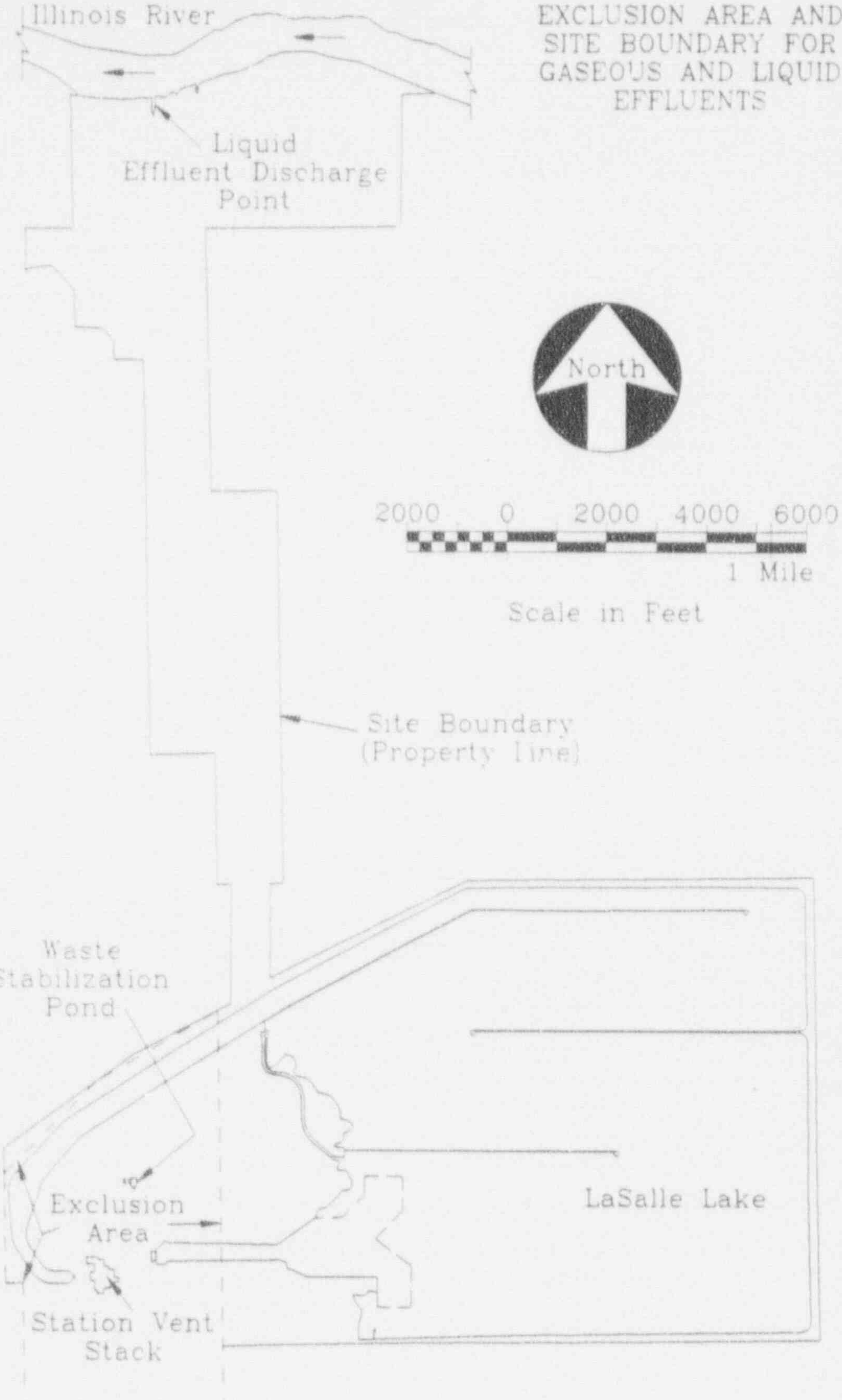
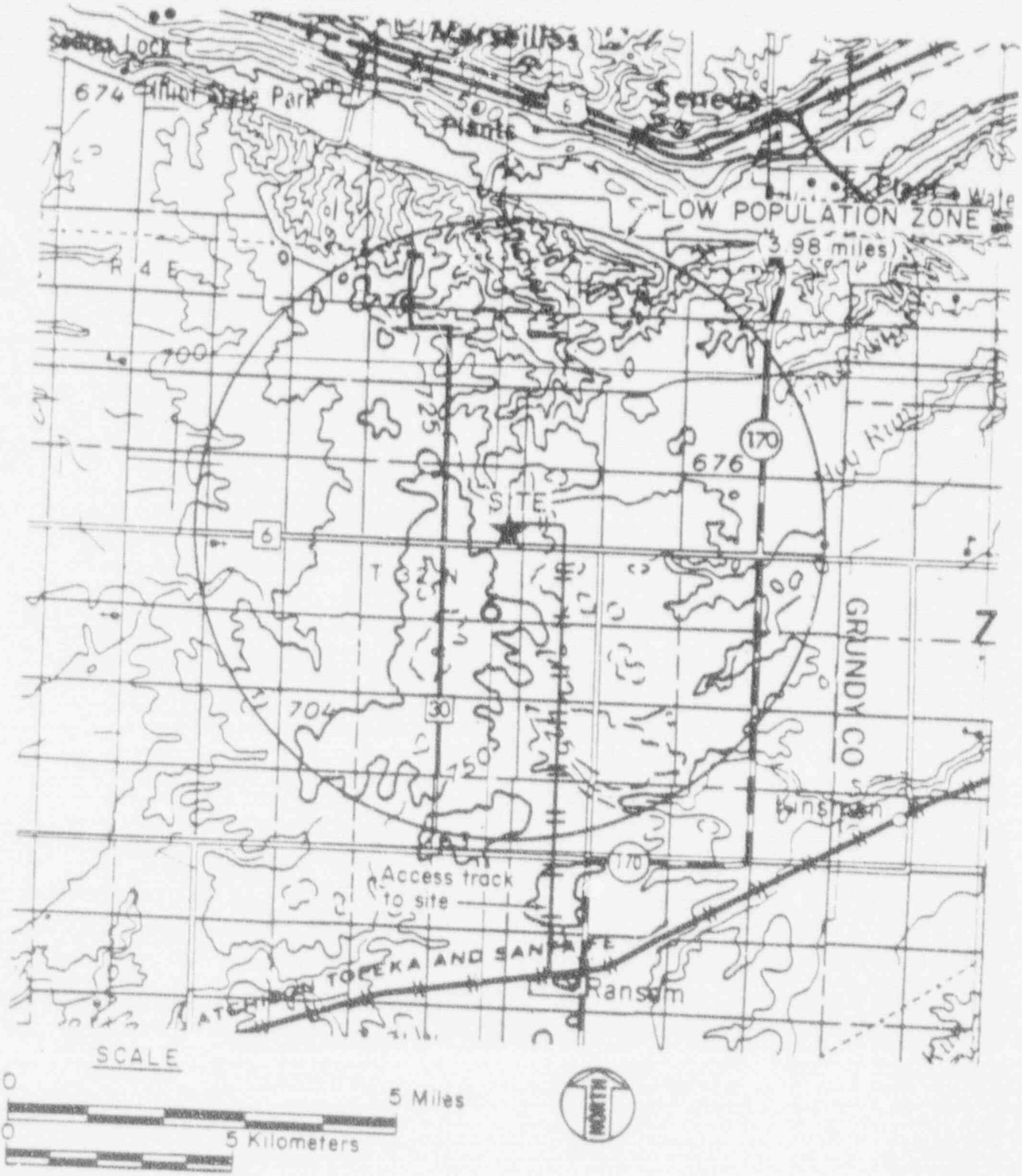

EXCLUSION AREA AND
SITE BOUNDARY FOR
GASEOUS AND LIQUID
EFFLUENTS
Illinois River
Liquid
Effluent Discharge
Point
North
2000 0 2000 4000 6000
1 Mile
Scale in Feet
Site Boundary
(Property line)
Waste
Stabilization
Pond
Exclusion
Area
Station Vent
Stack
LaSalle Lake

Figure 5.1.1-1



LOW POPULATION ZONE

Figure 5.1.2-1

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

F. The following programs shall be established, implemented, and maintained:

1. Primary Coolant Sources Outside Primary Containment

A program to reduce leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

2. In-Plant Radiation Monitoring

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

a. Training of personnel,

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

3. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis,

c. Provisions for maintenance of sampling and analysis equipment.

4. 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:

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

a. 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,

b. 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,

c. 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,

d. 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,

e. 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,

f. 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,

g. 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,

h. Limitations on the annual and 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,

i. 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,

j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

k. 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.

5. 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:

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

b. 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

c. 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.

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

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

b. Each REPORTABLE EVENT shall be reviewed pursuant to Specification 6.1.G.2.c(1).

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Vice President BWR Operations or his designated alternate. The incident shall be reviewed pursuant to Specifications 6.1.G.1.a and 6.1.G.2.a and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President BWR Operations and the Manager of Offsite Review and Investigative Function shall be notified within 24 hours.

6.5 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:

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

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

3. Records and reports of reportable events;

4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;

5. Records of changes to operating procedures;

6. Shift engineers' logs; and

7. Byproduct material inventory records and source leak test results.

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

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;

2. Changes made to the plant as it is described in the SAR;

3. Records of new and spent fuel inventory and assembly histories;

4. Updated, corrected, and as-built drawings of the plant;

5. Records of plant radiation and contamination surveys;

6. Records of offsite environmental monitoring surveys;

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

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

PLANT OPERATING RECORDS (Continued)

9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);
10. Records of individual staff members indicating qualifications, experience, training, and retraining;
11. Inservice inspections of the reactor coolant system;
12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;
13. Records of reactor tests and experiments;
14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;
15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;
16. Records of the service lives of all hydraulic and mechanical snubbers required by specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;
17. Records of analyses required by the radiological environmental monitoring program; and
18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.6 REPORTING REQUIREMENTS

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

A. Routine Reports

1. Startup Report

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

6.6 REPORTING REQUIREMENTS (Continued)

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

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

2. Annual Report

A tabulation shall be submitted on an annual basis prior to March 1 of each year 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 (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5 shall be included in the Annual Report along with the following information: (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) Clean-up 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.

3. Annual Radiological Environmental Operating Report*

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.

4. Semiannual Radioactive Effluent Release Report**

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.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Nuclear Reactor Regulation, Mail Station P1-137, US Nuclear Regulatory Commission, Washington, DC 20555, with a copy of the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.

6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

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

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

C. Unique Reporting Requirements

1. Special Reports shall be submitted to the Director of the Office of Inspection and Enforcement (Region III) within the time period specified for each report.

6.7 PROCESS CONTROL PROGRAM (PCP)*

6.7.1 The PCP shall be approved by the Commission prior to implementation.

6.7.2 Licensee initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. 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 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 Onsite Review and Investigative Function.

*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

6.8.1 The ODCM shall be approved by the Commission prior to implementation.

6.8.2 Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. 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 after review and acceptance by the On-Site Review and Investigative Function and the approval of the Plant Manager on the date specified by the On-Site Review and Investigative Function.

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 effective. 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.

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous and solid):

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;

3. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

*The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

4. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. NPF-18

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

   A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated May 22, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 set forth in 10 CFR Chapter I;

   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 enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 69 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

RJBarrett

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 1, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change. Pages indicated with an asterisk are provided for convenience.

REMOVE	INSERT
I	I
II	II
*III	*III
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DEFINITIONS

SECTION

1.0 DEFINITIONS

DEFINITIONS

SECTION

DEFINITIONS (Continued)

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

SECTION PAGE

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.11 RADIOACTIVE EFFLUENTS

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES (Continued)

LINEAR HEAT GENERATION RATE

1.21 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

1.22 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

1.23 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

1.24 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 licensee, 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

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.26 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 Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.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 Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.27 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, a normal and an emergency electrical power source, 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

1.28 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

1.29 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

1.30 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.31 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. Each primary containment air lock is OPERABLE pursuant to 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 OPERABLE pursuant to 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

1.32 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.33 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

1.34 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.35 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.

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.

SECONDARY CONTAINMENT INTEGRITY

1.38 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All 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, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

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

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

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

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.39 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

1.40 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

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.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.44 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

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

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.

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
P	Prior to each radioactive release.
N.A.	Not applicable.

3.3.7.10 deleted
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EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.11 The explosive gas monitoring instrumentation channels shown in Table 3.3.7.11-1 shall be OPERABLE with their Alarm/Trip setpoints set to ensure that the limits of specification 3.11.2.6 are not exceeded.

APPLICABILITY: During operation of the main condenser air ejector.

ACTION:

a. With an explosive gas 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 3.3.7.11-1.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3.7.11-1. Restore the inoperable instrumentation channels to an OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.6.C. within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.7.11 Each explosive gas monitoring instrumentation channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies shown in Table 4.3.7.11-1.

TABLE 3.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)		
a. Hydrogen Monitor	1/train	110

TABLE NOTATION

ACTION 110 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the 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. If the recombiner(s) temperature remains constant and THERMAL POWER has not changed, the grab sample collection frequency may be changed to 8 hours.

TABLE 4.3.7.11-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION*	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	D	M	Q	**

TABLE NOTATION

* The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

** During operation of the main condenser air ejector.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.12 The loos -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 6.6.c 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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.

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.8.1-1.

4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level-High, Level 8	3

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	< 55.5 inches*	< 56.0 inches*

*See Bases Figure B 3/4 3-1.

TABLE 4.3.8.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-High, Level 8	S	M	R

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 The quantity of radioactive material contained in any outside temporary tanks shall be limited to less than or equal to the limits calculated in the ODCM.

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.

ACTION:

a. With the concentration of hydrogen in the main condenser offgas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.1 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limits as required by Table 3.3.7.11-1 of Specification 3.3.7.11.

RADIOACTIVE EFFLUENTS

MAIN CONDENSER

LIMITING CONDITION FOR OPERATION

3.11.2.2 The release rate of the sum of the activities from the noble gases measured prior to the holdup line shall be limited to less than or equal to 3.4 x 105 microcuries/second.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the release rate of the sum of the activities of the noble gases prior to the holdup line exceeding 3.4 x 105 microcuries/second restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.11.2.7.1 The radioactivity rate of noble gases prior to the holdup line shall be continuously monitored in accordance with Specification 3.3.7.11.

4.11.2.7.2 The release rate of the sum of the activities from noble gases prior to the holdup line shall be determined to be within the limits of Specification 3.11.2.7 at the following frequencies by performing an isotopic analysis of a representative sample of gases taken prior to the holdup line.

a. At least once per 31 days.

b. Within 4 hours following an increase, as indicated by the off gas pre-treatment Noble Gas Activity Monitor, of greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

MONITORING INSTRUMENTATION (Continued)

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-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

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 should 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 (TIP) 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 specification allows use of substituted TIP data from symmetric channels if the control rod pattern is symmetric since the TIP data is adjusted by the plant computer to remove machine dependent and power level dependent bias. The source of data for the substitution may also be a 3-dimensional BWR core simulator calculated data set which is normalized to available real data. Since uncertainty could be introduced by the simulation and normalization process, an evaluation of the specific control rod pattern and core operating state must be performed to ensure that adequate margin to core operating limits is maintained.

3/4.3.7.8 DELETED

3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, increasing the frequency of fire watch patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.7.10 DELETED

3/4.3.7.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation provides for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

3/4.3.7.12 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."

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure to prevent overfilling the reactor vessel which may result in high pressure liquid discharge through the safety/relief valve discharge lines.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1. LIQUID EFFLUENTS

3/4.11.1.1 LIQUID HOLDUP TANKS

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 GASEOUS EFFLUENTS

3/4.11.2.1 EXPLOSIVE GAS MIXTURE

The specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits 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.

3/4.11.2.2 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.

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.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The primary containment is a steel lined post-tensioned concrete structure consisting of a drywell and suppression chamber. The drywell is a steel-lined post-stressed concrete vessel in the shape of a truncated cone closed by a steel dome. The drywell is above a cylindrical steel-lined post-stressed concrete suppression chamber and is attached to the suppression chamber through a series of downcomer vents. The drywell has a minimum free air volume of 229,538 cubic feet. The suppression chamber has an air region of 164,800 to 168,100 cubic feet and a water region of 128,800 to 131,900 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

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

a. Maximum internal pressure: 45 psig.

b. Maximum internal temperature: drywell 340°F.
suppression chamber 275°F.

c. Maximum external pressure: 5 psig.

d. Maximum floor differential pressure: 25 psid, downward.
5 psid, upward.

SECONDARY CONTAINMENT

5.2.3 The secondary containment consists of the Reactor Building, the equipment access structure and a portion of the main steam tunnel and has a minimum free volume of 2,875,000 cubic feet.

EXCLUSION AREA AND SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

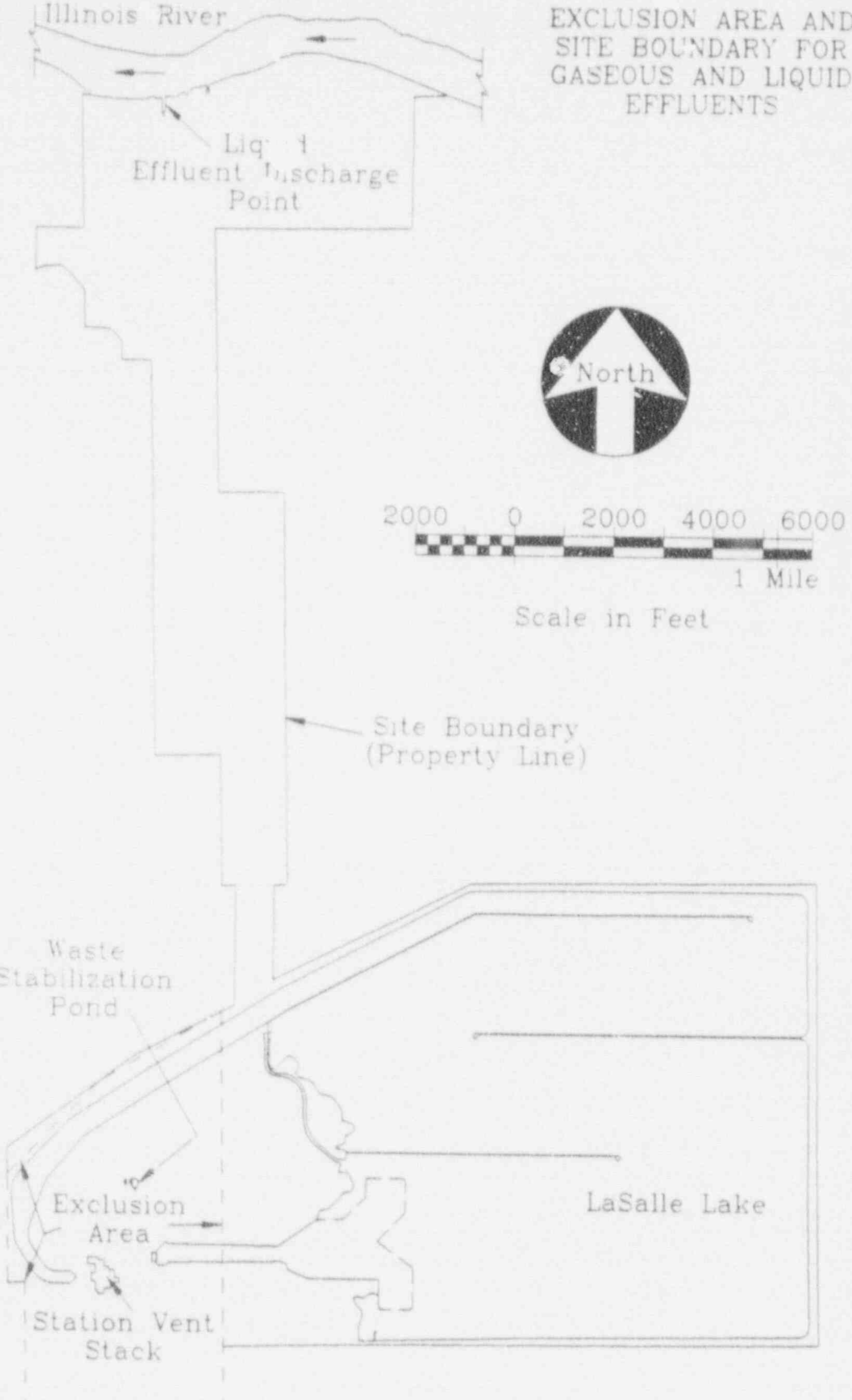
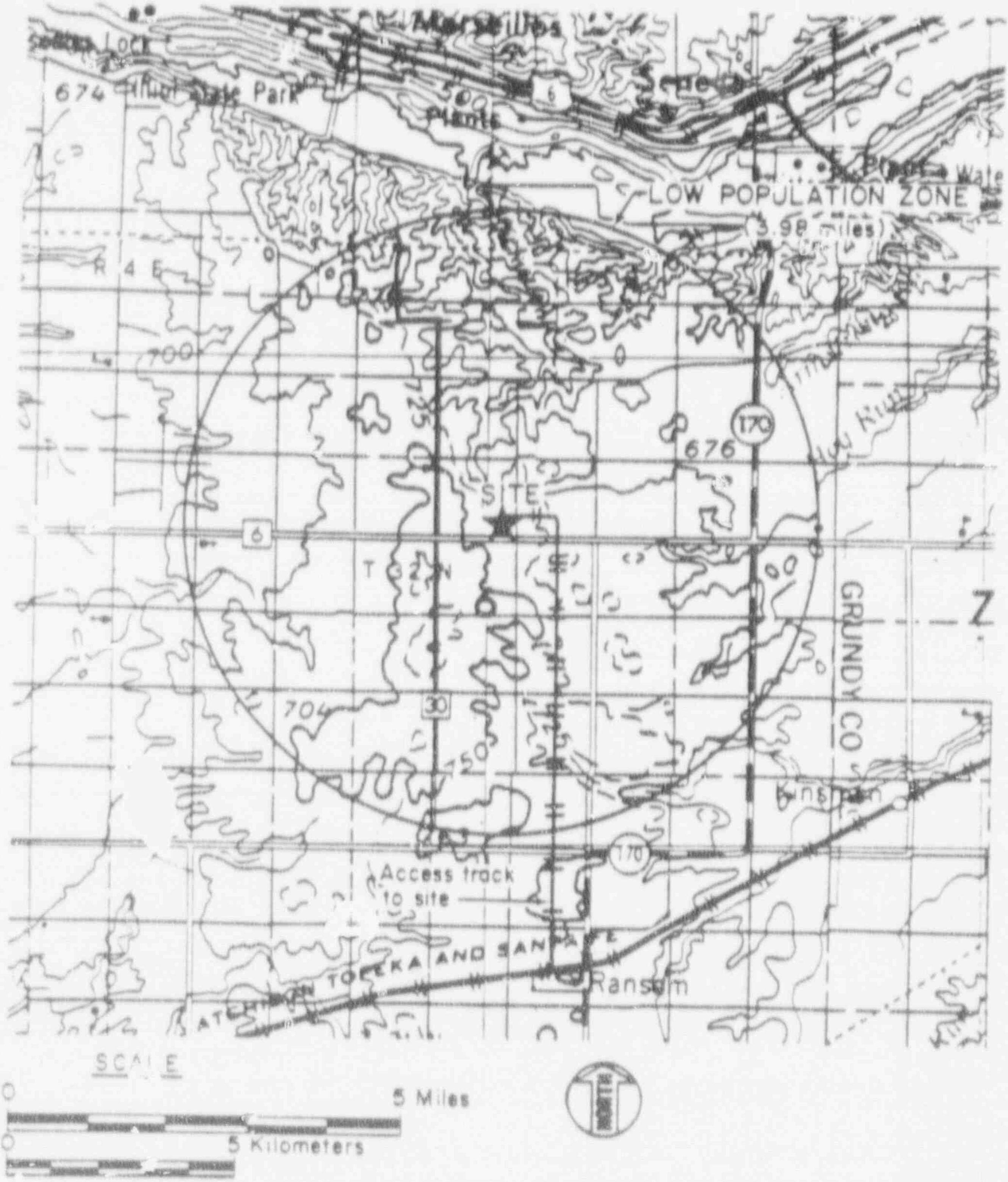


Figure 5.1.1-1



LOW POPULATION ZONE

Figure 5.1.2-1

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

F. The following programs shall be established, implemented, and maintained:

1. Primary Coolant Sources Outside Primary Containment

A program to reduce leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include LPCS, HPCS, RHR/LPCI, RCIC, hydrogen recombiner, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements, and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

2. In-Plant Radiation Monitoring

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

a. Training of personnel.

b. Procedures for monitoring, and

c. Provisions for maintenance of sampling and analysis equipment.

3. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel,

b. Procedures for sampling and analysis,

c. Provisions for maintenance of sampling and analysis equipment.

4. 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:

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

a. 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,

b. 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,

c. 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,

d. 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,

e. 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,

f. 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,

g. 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,

h. Limitations on the annual and 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,

i. 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,

j. Limitations on venting and purging of the containment through the Primary Containment Vent and Purge System or Standby Gas Treatment System to maintain releases as low as reasonably achievable,

k. 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.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

5. 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:

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

b. 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

c. 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.

6.3 ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE EVENT IN PLANT OPERATION

The following actions shall be taken for REPORTABLE EVENTS:

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

b. Each REPORTABLE EVENT shall be reviewed pursuant to Specification 6.1.G.2.c(1).

6.4 ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED

If a safety limit is exceeded, the reactor shall be shut down immediately pursuant to Specification 2.1.1, 2.1.2 and 2.1.3, and critical reactor operation shall not be resumed until authorized by the NRC. The conditions of shutdown shall be promptly reported to the Vice President BWR Operations or his designated alternate. The incident shall be reviewed pursuant to Specifications 6.1.G.1.a and 6.1.G.2.a and a separate Licensee Event Report for each occurrence shall be prepared in accordance with Section 50.73 to 10 CFR Part 50. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President BWR Operations and the Manager of Off-site Review and Investigative Function shall be notified within 24 hours.

6.5 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least 5 years:

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

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

3. Records and reports of reportable events;

4. Records and periodic checks, inspection and/or calibrations performed to verify that the surveillance requirements (see Section 4 of these specifications) are being met. All equipment failing to meet surveillance requirements and the corrective action taken shall be recorded;

5. Records of changes to operating procedures;

6. Shift engineers' logs; and

7. Byproduct material inventory records and source leak test results.

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

1. Substitution or replacement of principal items of equipment pertaining to nuclear safety;
2. Changes made to the plant as it is described in the SAR;
3. Records of new and spent fuel inventory and assembly histories;
4. Updated, corrected, and as-built drawings of the plant;
5. Records of plant radiation and contamination surveys;
6. Records of offsite environmental monitoring surveys;
7. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant, in accordance with 10 CFR Part 20;
8. Records of radioactivity in liquid and gaseous wastes released to the environment;
9. Records of transient or operational cycling for those components that have been designed to operate safety for a limited number of transient or operational cycles (identified in Table 5.7.1-1);

PLANT OPERATING RECORDS (Continued)

10. Records of individual staff members indicating qualifications, experience, training, and retraining;
11. Inservice inspections of the reactor coolant system;
12. Minutes of meetings and results of reviews and audits performed by the offsite and onsite review and audit functions;
13. Records of reactor tests and experiments:
14. Records of Quality Assurance activities required by the QA Manual, except for those items specified in Section 6.5.A;
15. Records of reviews performed for changes made to procedures on equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;
16. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records;
17. Records of analyses required by the radiological environmental monitoring program; and
18. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.6 REPORTING REQUIREMENTS

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

A. Routine Reports

1. Startup Report

A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.6 REPORTING REQUIREMENTS (Continued)

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

2. Annual Report

A tabulation shall be submitted on an annual basis prior to March 1 of each year 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 (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.5 shall be included in the Annual Report along with the following information: (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) Clean-up 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.

3. Annual Radiological Environmental Operating Report*

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.

4. Semiannual Radioactive Effluent Release Report**

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.

5. Monthly Operating Report

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Nuclear Reactor Regulation, Mail Station P1-137, US Nuclear Regulatory Commission, Washington, DC 20555, with a copy of the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by Onsite Review and Investigative Function.

6. Core Operating Limits Report

a. Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

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

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

C. Unique Reporting Requirements

1. Special Reports shall be submitted to the Director of the Office of Inspection and Enforcement (Region III) within the time period specified for each report.

6.7 PROCESS CONTROL PROGRAM (PCP)*

6.7.1 The PCP shall be approved by the Commission prior to implementation.

6.7.2 Licensee initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. 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 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 Onsite Review and Investigative Function.

*The Process Control Program (PCP) is common to La Salle Unit 1 and La Salle Unit 2.

6.8 OFFSITE DOSE CALCULATION MANUAL (ODCM)*

6.8.1 The ODCM shall be approved by the Commission prior to implementation.

6.8.2 Licensee-initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. 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 after review and acceptance by the On-Site Review and Investigative Function and the approval of the Plant Manager on the date specified by the On-Site Review and Investigative Function.

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 effective. 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.

6.9 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

6.9.1 Licensee initiated major changes to the radioactive waste treatment systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the Onsite Review and Investigative Function. The discussion of each change shall contain:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2. Sufficient detailed information to totally support the reason for the change without benefit or additional or supplemental information;

3. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

*The OFFSITE DOSE CALCULATION MANUAL (ODCM) is common to La Salle Unit 1 and La Salle Unit 2.

MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Continued)

4. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review and Investigative Function.

b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.