ATTACHMENT B
PAGES REVISED DUE TO PROPOSED AMENDMENT
TO LICENSE/TECHNICAL SPECIFICATIONS

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DEFINITIONS

SECTION

DESIGN FEATURES

ADMINISTRATIVE CONTROLS

LIST OF FIGURES

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CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

INSERT "B"

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit

MOVE TO NEW PAGE 1-2a

INSERT A

CONTROLLED AREA

1.6.a The CONTROLLED AREA shall be an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

INSERT B

DOSE EQUIVALENT

1.9.a DEEP DOSE EQUIVALENT, which applies to external whole-body exposure, shall be the DOSE EQUIVALENT at a tissue depth of 1 cm (1000 mg/cm^2).

I.9.b DOSE EQUIVALENT shall be the product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The unit of DOSE EQUIVALENT is the rem.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

INSERT "C"

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

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

LIMITING CONTROL ROD PATTERN

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

MOVE TO PAGE 1-4

INSERT C

HIGH RADIATION AREA

1.17.a A HIGH RADIATION AREA shall be an area, accessible to individuals, in which radiation levels could result in an individual receiving a DOSE EQUIVALENT in excess of 100 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

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

INSERT "D"

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.

INSERT "E"

INSERT D

MEMBER OF THE PUBLIC

1.24 A MEMBER OF THE PUBLIC shall be an individual in a CONTROLLED or UNRESTRICTED AREA. An individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose.

INSERT E

OCCUPATIONAL DOSE

1.25a OCCUPATIONAL DOSE is the dose received by an individual in a RESTRICTED AREA or in the course of employment in which the individual's assigned duties involve exposure to radiation and to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person.

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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.

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INSERT "F"

INSERT F

RESTRICTED AREA

1.36.a A RESTRICTED AREA shall be an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREAS do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

1.37 From page 1-6

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 i 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. INSERT "G"

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.

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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.

INSERT G

TOTAL EFFECTIVE DOSE EQUIVALENT

1.43.a TOTAL EFFECTIVE DOSE EQUIVALENT shall be the sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

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. Deleted

LOW POPULATION ZONE

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

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1. the Offsite Dose Calculation Manual.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1. the Offsite Dose Calculation Manual.

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.

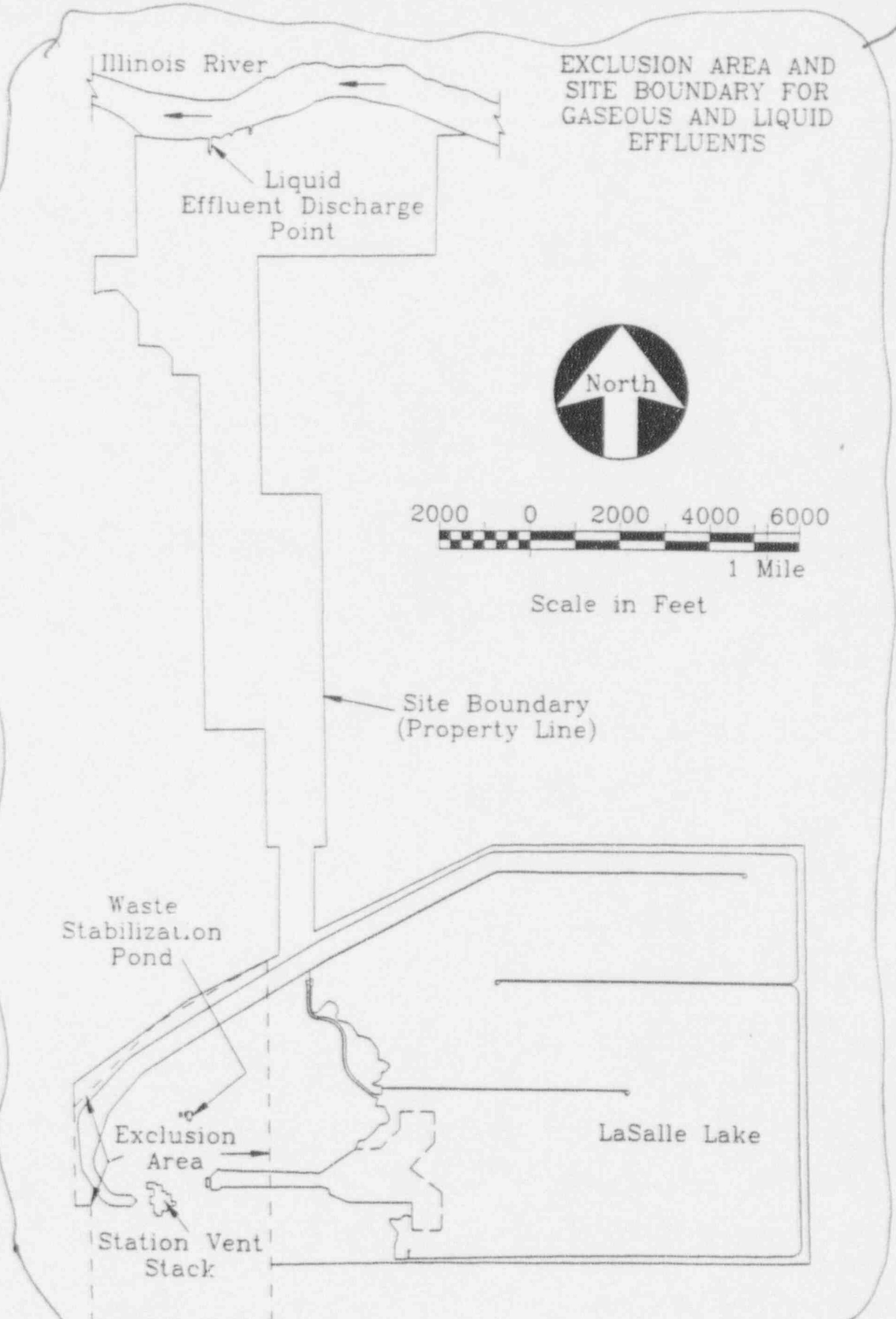
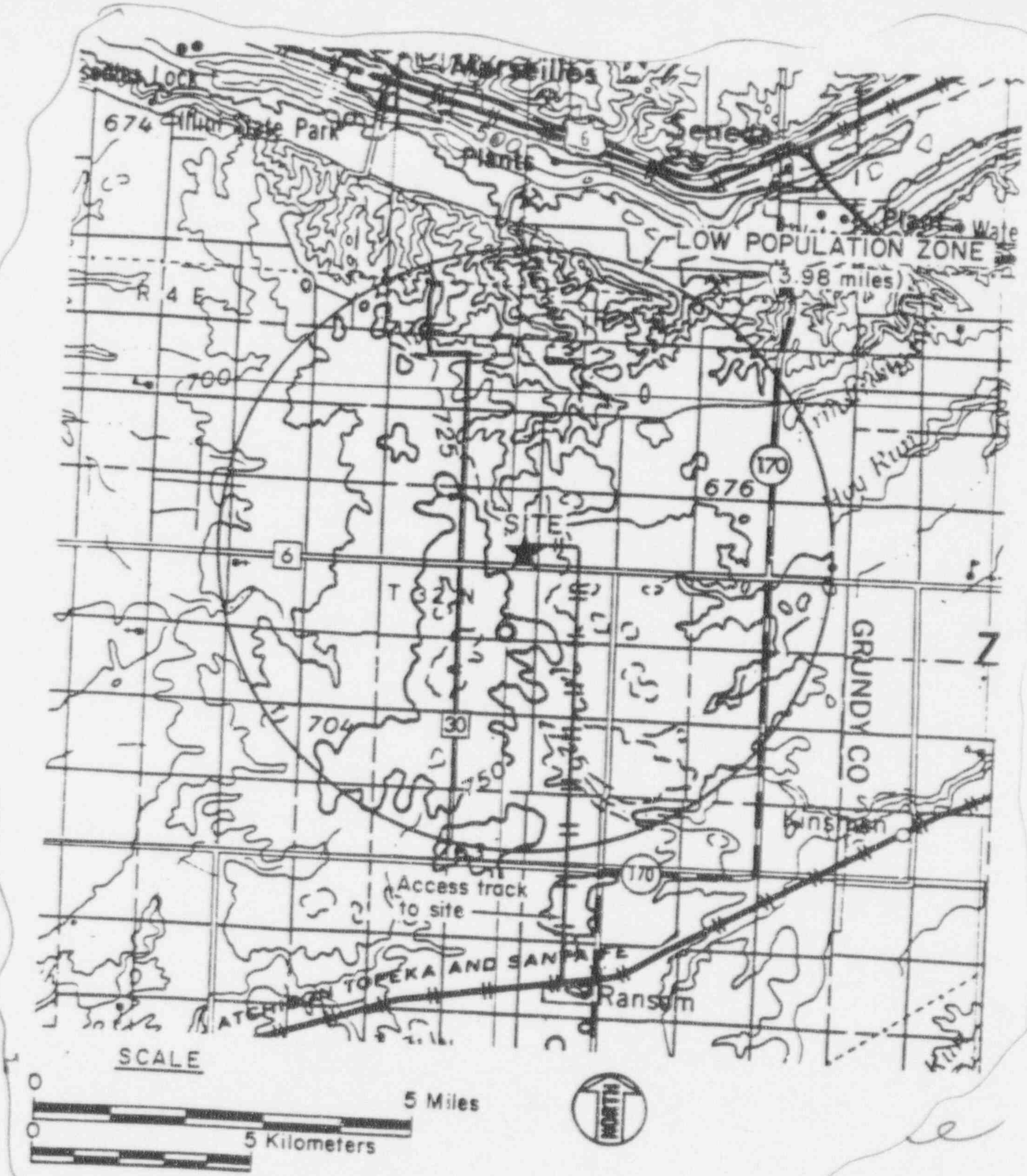

Illinois River
EXCLUSION AREA AND SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS
Liquid Effluent Discharge Point
North
2000 0 2000 4000 6000
1 Mile
Scale in Feet
Site Boundary (Property Line)
Waste Stabilization Pond
Exclusion Area
Station Vent Stack
LaSalle Lake

Figure 5.1.1-1



LOW POPULATION ZONE

Figure 5.1.2-1

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircalloy clad fuel rods with an initial composition of slightly enriched uranium dioxide, UO_2. Fuel assemblies shall be limited to those fuel designs approved for use in BWR's.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide power (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.
2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION Deleted

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

INSERT "H"

6.1.1 HIGH RADIATION AREAS

6.1.1.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr*, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

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

6.1.1.2 In addition to the requirements of 6.1.1.1, above, for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem*, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such High-High Radiation Areas for the time that access is required.

6.1.1.3 Keys to manually open computer controlled High Radiation Area doors and High-High Radiation Area doors shall be maintained under the Administration control of the Shift Supervisor on duty and/or the Health Physicist.

6.1.1.4 High-High Radiation areas, as defined in 6.1.1.2 above, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR 20.203 c.2 (iii), locked except during periods when access to the area is required with positive control over each individual entry, or 10 CFR 20.203.c.4. In the case of a High Radiation Area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within

*Measurement made at 18" from source of radioactivity.

HIGH RADIATION AREAS (Continued)

individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.2 PLANT OPERATING PROCEDURES AND PROGRAMS

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

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

   b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No 82-33,

   c. Station Security Plan implementation,

   d. Generating Station Emergency Response Plan implementation,

   e. PROCESS CONTROL PROGRAM implementation,

   f. OFFSITE DOSE CALCULATION MANUAL implementation, and

   g. Fire Protection Program implementation.

*Measurement made at 18" from source of radioactivity.

INSERT H

6.1.1 HIGH RADIATION AREA

6.1.1.1 Pursuant to Paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a), each high radiation area in which the dose rate is equal to or less than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by a Radiation Work Permit (RWP). (Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from RWP requirements during the performance of their assigned duties in high radiation areas in which the dose rate is less than or equal to 1000 mrem/h at 30 cm (12 in.), provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas). Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who shall perform periodic radiation surveillance at the frequency specified in the RWP.

6.1.1.2 a. In addition to the requirements of Specification 6.1.1.1, areas accessible to personnel with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. The computer can be programmed to permit entry through locked doors for any individual requiring access to any such high radiation areas. For high radiation areas greater than 1000 mrem/h at 30 cm (12 in.) <u>NOT</u> equipped with or having non-functional computerized card readers, doors shall be locked to prevent unauthorized entry. Keys to manually open all high radiation area doors shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision.

INSERT H
(continued)

2. Personnel access and exposure control over activities being performed within these areas shall be specified by an approved RWP. Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from RWP requirements during the performance of their assigned duties in high radiation areas in which the dose rate is greater than 1000 mrem/h at 30 cm (12 in.), provided they are otherwise following radiation protection procedures for entry into such high radiation areas.

3. Each person entering the area shall be provided with an alarming radiation monitoring device which continuously integrates the radiation dose rate (such as an electronic dosimeter). Continuous surveillance by a radiation protection technician may be substituted for an alarming dosimeter.

b. For individual areas accessible to personnel with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device.

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PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Station Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2 and 4.4, and be approved by the Station Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation in accordance with Specification 6.1.G.2.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

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D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.
2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
3. The change is documented, reviewed and approved in accordance with Specification 6.2.C. within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

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.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

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:

a. Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and set-point 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,
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,
20.1301

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,
2

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,

a MEMBER OF THE PUBLIC

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

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.

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).

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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;

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PLANT OPERATING RECORDS (Continued)

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);

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

6.6 REPORTING REQUIREMENTS (Continued)

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.

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.

Operating

2. Annual Report

Deep Dose Equivalent

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.

person-

Primary or secondary dosimetry

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**

calendar year

The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months of operation shall

Prior to April 1

~~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.

* 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.

Monthly Operating

Semiannual Radioactive Effluent Release Report (Continued)

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~~ A 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:

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The minimum Critical Power Ratio (MCPR) (including 20% scram time, tau (t), dependent MCPR limits, and K_f core flow MCPR adjustment factors) for Technical Specification 3.2.3.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For LaSalle County Station Unit 1, the topical reports are:

(1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

(2) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(3) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(4) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

INSERT "I"

B. Deleted

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.

INSERT I

6.6.A.7 Annual Reports of Individual Monitoring

The dose equivalent results of all individuals monitored for radiation exposure during the previous calendar year shall be submitted to the NRC prior to May 1 of each year. The results shall be submitted on an NRC Form 5 or electronic media containing all the information required by NRC Form 5.

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. 20.1301,

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.

INSERT "J"

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.

INSERT J

If the Commission holds a controlled copy of the ODCM, then the Licensee can submit only the revised ODCM pages.

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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.

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.

2

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.

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DEFINITIONS

SECTION

DEFINITIONS (Continued)

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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is tested.

INSERT "A"

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.8 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6.A.6. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.9 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the GEXL correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

INSERT "B"

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to energization of the recirculation pump circuit breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

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INSERT A

CONTROLLED AREA

1.6.a The CONTROLLED AREA shall be an area, outside of a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

INSERT D

MEMBER OF THE PUBLIC

1.24 A MEMBER OF THE PUBLIC shall be an individual in a CONTROLLED or UNRESTRICTED AREA. An individual is not a MEMBER OF THE PUBLIC during any period in which the individual receives an occupational dose.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

e

FRACTION OF LIMITING POWER DENSITY

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

FRACTION OF RATED THERMAL POWER

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

INSERT "C"

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

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

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

ISOLATION SYSTEM RESPONSE TIME

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

LIMITING CONTROL ROD PATTERN

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

INSERT B

DOSE EQUIVALENT

1.9.a DEEP DOSE EQUIVALENT, which applies to external whole-body exposure, shall be the DOSE EQUIVALENT at a tissue depth of 1 cm (1000 mg/cm^2).

I.9.b DOSE EQUIVALENT shall be the product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. The unit of DOSE EQUIVALENT is the rem.

INSERT C

HIGH RADIATION AREA

1.17.a A HIGH RADIATION AREA shall be an area, accessible to individuals, in which radiation levels could result in an individual receiving a DOSE EQUIVALENT in excess of 100 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

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.

INSERT "D"

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.

INSERT "E"

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 Operatirg 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).

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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.

INSERT E

OCCUPATIONAL DOSE

1.25a OCCUPATIONAL DOSE is the dose received by an individual in a RESTRICTED AREA or in the course of employment in which the individual's assigned duties involve exposure to radiation and to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person.

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.

INSERT "F"

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.

Move to new page 1-6a

INSERT F

RESTRICTED AREA

1.36.a A RESTRICTED AREA shall be an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. RESTRICTED AREAS do not include areas used as residential quarters, but separate rooms in a residential building may be set apart as a restricted area.

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.

INSERT "G"

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.

Move to new page 1-7a

INSERT G

TOTAL EFFECTIVE DOSE EQUIVALENT

1.43.a TOTAL EFFECTIVE DOSE EQUIVALENT shall be the sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

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. Deleted

LOW POPULATION ZONE

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

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1.1-1.
the OFFSITE Dose Calculation Manual.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1.1-1.
the OFFSITE Dose Calculation Manual.

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.

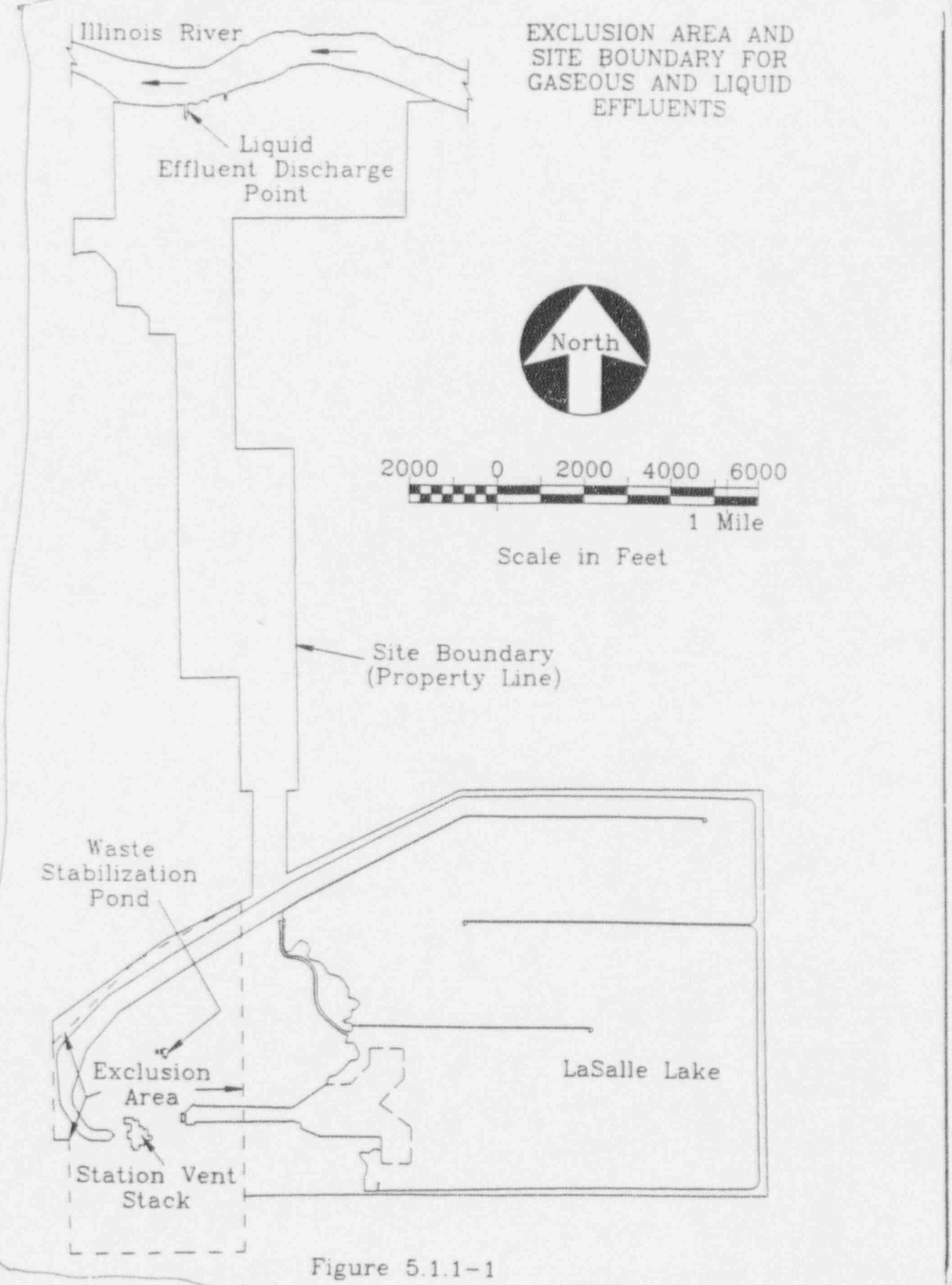
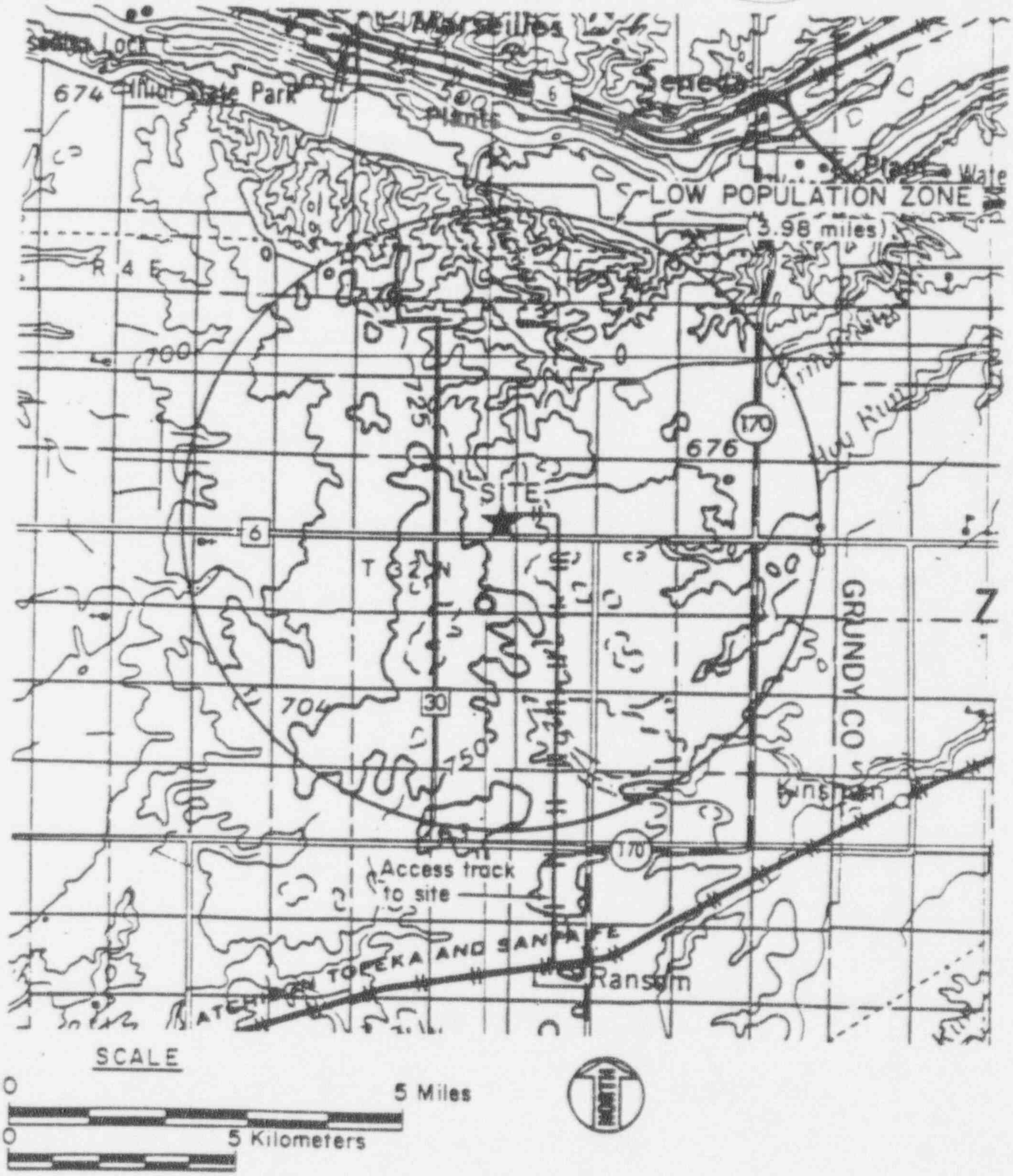

Illinois River
EXCLUSION AREA AND SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS
Liquid Effluent Discharge Point
North
2000 0 2000 4000 6000
1 Mile
Scale in Feet
Site Boundary (Property Line)
Waste Stabilization Pond
Exclusion Area
Station Vent Stack
LaSalle Lake

Figure 5.1.1-1



LOW POPULATION ZONE

Figure 5.1.2-1

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 764 fuel assemblies. Each assembly consists of a matrix of Zircalloy clad fuel rods with an initial composition of slightly enriched uranium dioxide, UO_2. Fuel assemblies shall be limited to those fuel designs approved for use in BWR's.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 185 cruicform shaped control rod assemblies. The control material shall be boron carbide powder (B_4C) and/or hafnium metal. The control rod assembly shall have a nominal axial absorber length of 143 inches.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pumps.

2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1500 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is ~ 21,000 cubic feet at a nominal T_{ave} of 533°F.

5.5 METEOROLOGICAL TOWER LOCATION

Deleted

5.5.1 The meteorological tower shall be located as shown on Figure 5.1.1-1.

6.1.1 HIGH RADIATION AREAS

6.1.1.1 Pursuant to Paragraph 20.203(c)(5) of 10 CFR 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr*, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

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

6.1.1.2 In addition to the requirements of 6.1.1.1, above, for areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem*, the computer shall be programmed to permit entry through locked doors for any individual requiring access to any such High-High Radiation Areas for the time that access is required.

6.1.1.3 Keys to manually open computer controlled High Radiation Area doors and High-High Radiation Area doors shall be maintained under the Administration control of the Shift Supervisor on duty and/or the Health Physicist.

6.1.1.4 High-High Radiation areas, as defined in 6.1.1.2 above, not equipped with the computerized card readers shall be maintained in accordance with 10 CFR 20.203 c.2 (iii), locked except during periods when access to the area is required with positive control over each individual entry, or 10 CFR 20.203.c.4. In the case of a High Radiation Area established for a period of 30 days or less, direct surveillance to prevent unauthorized entry may be substituted. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For

INSERT "H"

HIGH RADIATION AREAS (Continued)

individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem* that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote, such as use of closed circuit TV cameras, continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.2 PLANT OPERATING PROCEDURES AND PROGRAMS

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

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

   b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

   c. Station Security Plan implementation,

   d. Generating Station Emergency Response Plan implementation,

   e. PROCESS CONTROL PROGRAM implementation,

   f. OFFSITE DOSE CALCULATION MANUAL implementation, and

   g. Fire Protection Program implementation.

*Measurement made at 18" from source of radioactivity.

INSERT H

6.1.1 HIGH RADIATION AREA

6.1.1.1 Pursuant to Paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a), each high radiation area in which the dose rate is equal to or less than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by a Radiation Work Permit (RWP). (Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from RWP requirements during the performance of their assigned duties in high radiation areas in which the dose rate is less than or equal tc 1000 mrem/h at 30 cm (12 in.), provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas). Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and who shall perform periodic radiation surveillance at the frequency specified in the RWP.

6.1.1.2 a. In addition to the requirements of Specification 6.1.1.1, areas accessible to personnel with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. The computer can be programmed to permit entry through locked doors for any individual requiring access to any such high radiation areas. For high radiation areas greater than 1000 mrem/h at 30 cm (12 in.) NOT equipped with or having non-functional computerized card readers, doors shall be locked to prevent unauthorized entry. Keys to manually open all high radiation area doors shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision.

INSERT H
(continued)

2. Personnel access and exposure control over activities being performed within these areas shall be specified by an approved RWP. Individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals may be exempt from RWP requirements during the performance of their assigned duties in high radiation areas in which the dose rate is greater than 1000 mrem/h at 30 cm (12 in.), provided they are otherwise following radiation protection procedures for entry into such high radiation areas.

3. Each person entering the area shall be provided with an alarming radiation monitoring device which continuously integrates the radiation dose rate (such as an electronic dosimeter). Continuous surveillance by a radiation protection technician may be substituted for an alarming dosimeter.

b. For individual areas accessible to personnel with radiation levels greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas, such as the containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be roped off, conspicuously posted, and a flashing light shall be activated as a warning device.

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PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

B. Radiation control procedures shall be maintained, made available to all station personnel, and adhered to. These procedures shall show permissible radiation exposure and shall be consistent with the requirements of 10 CFR 20. This radiation protection program shall be organized to meet the requirements of 10 CFR 20.

C. TECHNICAL REVIEW AND CONTROL

Procedures required by Specification 6.2.A and 6.2.B and other procedures which affect nuclear safety, as determined by the Station Manager, and changes thereto, other than editorial or typographical changes, shall be reviewed as follows prior to implementation except as noted in Specification 6.2.D:

1. Each procedure or procedure change shall be independently reviewed by a qualified individual knowledgeable in the area affected other than the individual who prepared the procedure or procedure change. This review shall include a determination of whether or not additional cross-disciplinary reviews are necessary. If deemed necessary, the reviews shall be performed by the qualified review personnel of the appropriate discipline(s).

2. Individuals performing these reviews shall meet the applicable experience requirements of ANSI N18.1-1971, Sections 4.2 and 4.4, and be approved by the Station Manager.

3. Applicable Administrative Procedures recommended by Regulatory Guide 1.33, Plant Emergency Operating Procedures, and changes thereto shall be submitted to the Onsite Review and Investigative Function for review and approval prior to implementation in accordance with Specification 6.1.G.2.

4. Review of the procedure or procedure change will include a determination of whether or not an unreviewed safety question is involved. This determination will be based on the review of a written safety evaluation prepared by a qualified individual or documentation that a safety evaluation is not required. Onsite Review, Offsite Review and Commission approval of items involving unreviewed safety questions shall be obtained prior to Station approval for implementation.

5. The Department Head approval authority shall be specified in station procedures.

6. Written records of reviews performed in accordance with this specification shall be prepared and maintained in accordance with Specification 6.5.

7. Editorial and Typographical changes shall be made in accordance with station procedures.

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D. Temporary changes to procedures 6.2.A and 6.2.B above may be made provided:

1. The intent of the original procedure is not altered.
2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
3. The change is documented, reviewed and approved in accordance with Specification 6.2.C. within 14 days of implementation.

E. Drills of the emergency procedures described in Specification 6.2.A.4 shall be conducted at frequencies as specified in the Generating Stations Emergency Plan (GSEP). These drills will be planned so that during the course of the year, communication links are tested and outside agencies are contacted.

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.

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

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:

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,
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,
20.1301

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,
2

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,
a MEMBER OF THE PUBLIC

PLANT OPERATING PROCEDURES AND PROGRAMS (Continued)

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.

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).

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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;

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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 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.

Operating

Deep Dose Equivalent

2. Annual Report

person-

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.

Primary or Secondary dosimetry.

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.

prior to April 1

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.

A

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.

INSERT I

6.6.A.7 Annual Reports of Individual Monitoring

The dose equivalent results of all individuals monitored for radiation exposure during the previous calendar year shall be submitted to the NRC prior to May 1 of each year. The results shall be submitted on an NRC Form 5 or electronic media containing all the information required by NRC Form 5.

Core Operating Limits.

Semiannual Radioactive Effluent Release Report (Continued)

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The minimum Critical Power Ratio (MCPR) (including 20% scram time, tau (τ), dependent MCPR limits, and Kf core flow MCPR adjustment factors) for Technical Specification 3.2.3.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

(4) The Rod Block Monitor Upscale Instrumentation Setpoints for Technical Specification Table 3.3.6-2.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For LaSalle County Station Unit 2, the topical reports are:

(1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).

(2) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).

(3) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).

(4) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS Limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the U.S. Nuclear Regulatory Commission Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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B. Deleted.

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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.

   20.1301

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

INSERT "J"

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.

INSERT J

If the Commission holds a controlled copy of the ODCM, then the Licensee can submit only the revised ODCM pages.

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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.

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.

2

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.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS

Commonwealth Edison has evaluated this proposed amendment and determined that it involves no significant hazards considerations. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility is in accordance with the proposed amendment and would not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

The proposed amendment changes consists of: (1) adding definitions for: controlled area, dose equivalent, high radiation area, occupational dose, and total effective dose equivalent; revising the definition for member of the public; (2) relocating certain information in Section 5; (3) adding high radiation area controls; (4) extending the Radioactive Effluent Release Report submittal interval from semiannual to annual; and (5) editorial changes; and determined that it does not constitute a Significant Hazards Considerations.

1. The proposed changes do not involve a significant increase in the probability on consequences of an accident previously evaluated.

    The definitions and location of previous 10CFR20 requirements do not impact previously evaluated accidents because there is no change in the types and amounts of effluents that will be released. There will be no increase in individual or cumulative occupational radiation exposures. The proposed changes to the high radiation area controls exposure monitoring; will not adversely impact controls for exposure monitoring and do not effect the effluents or exposures.

    Relocating information from Section 5 to the Offsite Dose Calculation Manual (ODCM) and the editorial changes are administrative in nature. The proposed changes do not reduce any Technical Specification requirements. The changes provide consistency and improve readability. The information relocated from Section 5 is to be covered in more detail in the ODCM. The level of control is maintained.

    Changing the frequency of submitting the Radiological Effluent Release Report from semiannual to annual is consistent with the revised requirements of 10CFR50.36a. The change does not adversely impact the ability to meet applicable regulatory requirements related to liquid and gaseous effluents. The report is a historical record of station effluents and has no impact on the actual release process. The NRC will continue to receive the same information, but on an annual basis.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATIONS (CONTINUED)

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes are administrative in nature and do not affect plant design or operation. There is no change to the types and amounts of effluent that will be released, nor is there an increase in individual or cumulative occupational radiation exposures.

3. The proposed changes do not involve a significant reduction in a margin of safety.

   The proposed amendment does not involve the relaxation of any criteria identified in the SAR or reduce any of the requirements of Technical Specifications. The changes provide consistency and readability associated with the revised 10CFR20. There is no increase in the probability or consequences of an accident, and there is no impact on equipment important to safety or systems, structures or components because the changes are administrative in nature. There is no associated change to the type, amount, or control of radioactive effluents, nor is there an associated increase in individual or cumulative occupational radiation exposure.

   Therefore, based on the above evaluation, Commonwealth Edison has concluded that the changes to not involve Significant Hazards Consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT

Commonwealth Edison has evaluated the proposed amendment against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed change meets the criteria for a categorical exclusion as provided for under 10CFR51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10CFR50 and the amendment meets the following specific criteria:

(i) the amendment involves no significant hazards considerations

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards considerations.

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite

As documented in Attachment A, there will be no change in the types or significant increase in the amounts of any effluents released offsite.

(iii) there is no significant increase in individual or cumulative occupational radiation exposure

The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.