March 16, 1999

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M99548 AND M99549)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 131 to Facility Operating License No. NPF-11 and Amendment No. 116 to Facility Operating License No. NPF-18 for the LaSalle County Station, Units 1 and 2, respectively. The amendments are in response to your application dated August 14, 1998, as supplemented by letters dated October 13 and December 23, 1998.

The amendments revise the Technical Specifications (TSs) to reflect the use of Siemens Power Corporation ATRIUM-9B fuel. Specifically, the amendments incorporate the following into the TSs: (1) new methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates; (2) administrative changes that adopt improved Standard Technical Specification (iSTS) language where appropriate; and (3) changes to the Minimum Critical Power Ratio (MCPR).

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

Sincerely,

ORIG. SIGNED BY

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No.131to NPF-11
2. Amendment No.116to NPF-18
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 16, 1999

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Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
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Sincerely,

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures:
1. Amendment No.131 to NPF-11
2. Amendment No.116 to NPF-18
3. Safety Evaluation

cc w/encls: See next page

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 131
License No. NPF-11

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

   A. The application for amendment filed by Commonwealth Edison Company (the licensee), dated August 14, 1998, as supplemented on October 13 and December 23, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

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(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 131, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup of Cycle 9.

FOR THE NUCLEAR REGULATORY COMMISSION

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 16, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 131

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

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

REMOVE	INSERT
I	I
1-1	1-1
2-1	2-1
B 2-2	B 2-2
B 2-3	B 2-3
3/4 2-1	3/4 2-1
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
6-25b	6-25b

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS PAGE

1.1 ACTION 1-1

1.2 DELETED 1-1

1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE 1-1

1.4 CHANNEL CALIBRATION 1-1

1.5 CHANNEL CHECK 1-1

1.6 CHANNEL FUNCTIONAL TEST 1-1

1.7 CORE ALTERATION 1-2

1.8 CORE OPERATING LIMITS REPORT 1-2

1.9 CRITICAL POWER RATIO 1-2

1.10 DOSE EQUIVALENT I-131 1-2

1.11 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY 1-2

1.12 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME 1-2

1.13 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME 1-2

1.14 DELETED 1-3

1.15 FRACTION OF RATED THERMAL POWER 1-3

1.16 FREQUENCY NOTATION 1-3

1.17 GASEOUS RADWASTE TREATMENT SYSTEM 1-3

1.18 IDENTIFIED LEAKAGE 1-3

1.19 ISOLATION SYSTEM RESPONSE TIME 1-3

1.20 DELETED 1-4

1.21 LIMITING CONTROL ROD PATTERN 1-4

1.22 LINEAR HEAT GENERATION RATE 1-4

1.23 LOGIC SYSTEM FUNCTIONAL TEST 1-4

1.24 DELETED 1-4

1.25 MEMBER(S) OF THE PUBLIC 1-4

1.26 MINIMUM CRITICAL POWER RATIO 1-4

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.

1.2 DELETED

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 within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping or total channel steps so 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.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with two recirculation loop operation and shall not be less than 1.09 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.08 with two recirculation loop operation or less than 1.09 with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.4.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-7. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation, ANF-1125(P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

2.1.2 THERMAL POWER, High Pressure and High Flow (Continued)

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.4 LINEAR HEAT GENERATION RATE

GE Fuel

The specification assures that the LINEAR HEAT GENERATION RATE (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated. The effects of fuel densification are discussed in the General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A. The GESTAR discusses the methods used to ensure LHGR remains below the design limit.

SPC Fuel

The Linear Heat Generation Rate (LHGR) is a measure of the heat generation rate per unit length of a fuel rod in a fuel assembly at any axial location. LHGR limits are specified to ensure that fuel integrity limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). Operation above the LHGR limit followed by the occurrence of an AOO could potentially result in fuel damage and subsequent release of radioactive material. Sustained operation in excess of the LHGR limit could also result in exceeding the fuel design limits. The failure mechanism prevented by the LHGR limit that could cause fuel damage during AOOs is rupture of the fuel rod cladding caused by strain from the expansion of the fuel pellet. One percent plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. The $LHGRFAC_f$ multipliers are used to protect the core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ and $LHGRFAC_p$ multipliers are specified in the CORE OPERATING LIMITS REPORT.

References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993 and ANF-91-048(P)(A), Supplement 1 and Supplement 2, "BWR Jet Pump Model Revision for RELAX," October 1997.

3/4.2.4 LINEAR HEAT GENERATION RATE

References Con't:

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2, Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

6. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, Exxon Nuclear Company, February 1987.

7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

8. LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-32258P, General Electric Company, October 1993.

9. ARTS Improvement Program analysis for LaSalle County Station Units 1 and 2, NEDC-31531P, General Electric Company, December 1993.

Core Operating Limits Report (Continued)

(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

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

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

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

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

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-18

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

   A. The application for amendment filed by Commonwealth Edison Company (the licensee), dated August 14, 1998, as supplemented on October 13 and December 23, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 116, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to startup of Cycle 8.

FOR THE NUCLEAR REGULATORY COMMISSION

Donna M. Skay

Donna M. Skay, Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 16, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. NP -18

DOCKET NO. 50-374

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

REMOVE	INSERT
I	I
1-2	1-1
2-1	2-1
*2-2	*2-2
B 2-2	B 2-2
3/4 2-1	3/4 2-1
B 3/4 2-5	B 3/4 2-5
6-25a	6-25a
---	6-25b

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS PAGE

1.1 ACTION 1-1

1.2 DELETED 1-1

1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE 1-1

1.4 CHANNEL CALIBRATION 1-1

1.5 CHANNEL CHECK 1-1

1.6 CHANNEL FUNCTIONAL TEST 1-1

1.7 CORE ALTERATION 1-2

1.8 CORE OPERATING LIMITS REPORT 1-2

1.9 CRITICAL POWER RATIO 1-2

1.10 DOSE EQUIVALENT I-131 1-2

1.11 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY 1-2

1.12 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME 1-2

1.13 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME 1-2

1.14 DELETED 1-3

1.15 FRACTION OF RATED THERMAL POWER 1-3

1.16 FREQUENCY NOTATION 1-3

1.17 GASEOUS RADWASTE TREATMENT SYSTEM 1-3

1.18 IDENTIFIED LEAKAGE 1-3

1.19 ISOLATION SYSTEM RESPONSE TIME 1-3

1.20 DELETED 1-3

1.21 LIMITING CONTROL ROD PATTERN 1-4

1.22 LINEAR HEAT GENERATION RATE 1-4

1.23 LOGIC SYSTEM FUNCTIONAL TEST 1-4

1.24 DELETED 1-4

1.25 MEMBER(S) OF THE PUBLIC 1-4

1.26 MINIMUM CRITICAL POWER RATIO 1-4

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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.

1.2 DELETED

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 within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so 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.

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with two recirculation loop operation and shall not be less than 1.09 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.08 with two recirculation loop operation or less than 1.09 with single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.4.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.4.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the ANF Critical Power Methodology for boiling water reactors (Reference 1) which is a statistical model that combines all of the uncertainties in operation parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the SPC-developed ANFB critical power correlation.

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-7. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A), Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.

2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.

3. ANFB Critical Power Correlation, ANF-1125(P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.

4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

6. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

7. ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.4 LINEAR HEAT GENERATION RATE (Continued)

fuel damage caused by overstraining of the fuel cladding is not expected to occur. Fuel design evaluations are performed to demonstrate that the mechanical design limits are not exceeded during continuous operation with LHGRs up to the limit defined in the CORE OPERATING LIMITS REPORT. The analysis also includes allowances for short term transient operation above the LHGR limit.

At reduced power and flow conditions, the LHGR limit may need to be reduced to ensure adherence to the fuel mechanical design bases during limiting transients. At reduced power and flow conditions, the LHGR limit is reduced (multiplied) using the smaller of either the flow dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power. The $LHGRFAC_f$ multipliers are used to protect the core during slow flow runout transients. The $LHGRFAC_p$ multipliers are used to protect the core during plant transients other than core flow transients. The applicable $LHGRFAC_f$ and $LHGRFAC_p$ multipliers are specified in the CORE OPERATING LIMITS REPORT.

References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993 and "BWR Jet Pump Model Revision for RELAX," ANF-91-048(P)(A), Supplement 1 and Supplement 2, October 1997.

2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.

3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A) Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

Core Operating Limits Report (Continued)

(9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.

(10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.

(11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.

(12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.

(13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.

(14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.

(15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

(16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.

(17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.

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

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

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

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

Core Operating Limits Report (Continued)

(22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

(23) BWR Jet Pump Model Revision for RELAX, ANF-91-048(P)(A), Supplement 1 and Supplement 2, Siemens Power Corporation, October 1997.

(24) ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997.

(25) ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. NPF-11 AND

AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-18

COMMONWEALTH EDISON COMPANY

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated August 14, 1998, Commonwealth Edison Company (ComEd, the licensee) requested changes to the Technical Specifications (TSs) for LaSalle County Station, Units 1 and 2. This submittal superseded a submittal dated August 29, 1997, in its entirety. Additional information was submitted by letters dated October 13, 1998, and December 23, 1998. The December 23, 1998, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination. The proposed changes are designed to support the use of Siemens Power Corporation (SPC) ATRIUM-9B fuel during future operating cycles. The licensee proposed the following key changes:

1) incorporation of SPC's new methodologies that will enhance operational flexibility and reduce the likelihood of future plant derates.

2) changes to the LaSalle Minimum Critical Power Ratio (MCPR) Safety Limits, and

3) administrative changes that adopt improved Standard Technical Specification (iSTS) language where appropriate.

2.0 EVALUATION

The TS changes required to support the use of ATRIUM-9B fuel in LaSalle, Units 1 and 2, are evaluated below:

2.1 Addition of Topical Reports

The licensee proposes to add the following topical reports to section 6.6.A.6.b of the LaSalle TSs:

9903220114 990316
PDR ADOCK 05000373
P PDR

- BWR Jet Pump Model Revision for RELAX, ANF-91-0048(P)(A), Supplements 1 and 2, Siemens Power Corporation, October 1997;

- ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, Siemens Power Corporation, August 1997;

- ANFB Critical Power Correlation Determination of ATRIUM-9B Additive Constant Uncertainties, ANF-1125(P)(A), Supplement 1, Appendix E, Siemens Power Corporation, September 1998.

Addition of SPC Revised Jet Pump Methodology

The addition of the SPC Revised Jet Pump Methodology is needed to eliminate overly conservative average planar linear heat generation rate (APLHGR) limits derived from the currently referenced jet pump model. The Revised Jet Pump Methodology has been reviewed and approved by the NRC and is appropriate for the LaSalle units. Thus, addition of this methodology to the list of methodologies used for determining core operating limits is acceptable. Use of the revised jet pump methodology will continue to ensure that values for cycle specific parameters are determined such that all applicable limits of the safety analysis are met. Therefore, this change is acceptable.

Addition of Methodology for Co-resident Fuel

LaSalle, Units 1 and 2, will be transitioning from General Electric (GE) to SPC fuel, including the use of associated methodologies. Due to the transition to SPC fuel, it was necessary for SPC to provide a methodology for application of their ANFB critical power correlation to the co-resident GE fuel. This topical report has been reviewed and approved by the NRC. The approval of this report listed the following two conditions:

1) This methodology is applicable to once burned co-resident fuel. Lead assemblies are excluded.

2) A table comparing minimum critical power ratio (MCPR) data throughout the first reload exposure must be submitted to justify each plant application.

ComEd provided the information to address the conditions for Unit 2 in a letter dated March 8, 1996. By letter dated December 23, 1998, ComEd stated that it will comply to the above conditions for Units 1 and 2 and committed to provide the requested table for Unit 1 as soon as the core design for Unit 1, Cycle 9, is available. Since the conditions for implementation of the topical are required to be met by the inclusion of the approved version of the topical report EMF-1125(P)(A) in the TS, the use of SPC Generic Methodology for Application of ANFB Critical Power Correlation to Non-SPC Fuel, EMF-1125(P)(A), Supplement 1, Appendix C, is acceptable for Units 1 and 2. The addition of this methodology to the LaSalle, Units 1 and 2, TS will ensure that values of cycle-specific parameters are determined such that all applicable limits of the safety analysis are met.

Addition of SPC Topical

The NRC's approval of this topical listed the following three conditions:

1) The Additive Constant Uncertainty of 0.027 is applicable to SPC ATRIUM-9B fuel with a local peaking factor of up to and including 1.22.

2) For ATRIUM-9B fuel rods with a local peaking factor exceeding 1.22, with a maximum of the design limit specified on page 12 of Appendix E, an additional uncertainty will be imposed on a rod by rod basis such that the Additive Constant Uncertainty value of 0.029 will be used.

3) The additive constants and additive constant uncertainties described in Appendix E are applicable to ATRIUM-9B fuel operated within the following parameter ranges.

Pressure (psia)	600 to 1400
Mass Flow Rate (lb/s)	4.8 to 41.7
Inlet Subcooling (Btu/lb)	8 to 82

ComEd committed to these conditions for Units 1 and 2 in its December 23, 1998, letter and will be required to comply with these conditions by the inclusion of the approved version of the topical report in the TS which lists these conditions. Thus, the SPC topical for ATRIUM-9B fuel, ANF-1125(P)(A), Supplement 1, Appendix E, is acceptable for use for LaSalle, Units 1 and 2. The addition of this methodology will ensure that values for cycle-specific parameters are determined such that all applicable limits of the safety analysis are met.

2.2 Change to the MCPR Safety Limit

The licensee proposes to change TS 2.1.2 to revise the MCPR Safety Limit due to the transition to SPC fuel. Using the SPC ANFB Critical Power Correlation methodology and the ATRIUM-9B additive constant uncertainty resulting from the approval of ANF-1125(P)(A), Appendix E, the MCPR Safety Limit for the LaSalle units will be 1.08 for dual loop operation and 1.09 for single loop operation. These values bound LaSalle, Unit 2, Cycle 8, operation and it is expected that these values will be supported for LaSalle, Unit 1, Cycle 9, operation. The applicability of the MCPR Safety Limit will be confirmed on a cycle-specific basis. The values are anticipated to bound the actual MCPR Safety Limit for future LaSalle SPC reloads. Since the MCPR Safety Limits of 1.08 for dual loop operation and 1.09 for single loop operation were calculated with an approved methodology and use the approved additive constant uncertainty from Appendix E, the new values will ensure that 99.9 percent of the fuel rods will avoid transition boiling during normal operation and anticipated operational occurrence and are acceptable.

2.3 Revision to Thermal Limit Descriptions

The current Technical Specification 3.2.1 specifies the APLHGR limit as a function of the average planar exposure. However, the results of SPC's approved loss-of-coolant accident

(LOCA) methodology may be applied on either a bundle average or average planar exposure. Thus, the licensee proposed a less stringent description of APLHGR by deleting the reference to the average planar exposure from TS 3.2.1. This would allow the APLHGR limits to be based on either bundle average or average planar exposure. The detailed information to which the APLHGR is monitored will be specified in the Core Operating Limits Report (COLR). This generalization of the definition of APLHGR is consistent with the improved Standard Technical Specification (NUREG 1433/1434, Revision 1) wording. Both maximum APLHGRs (bundle average exposure based and planar average exposure based) are acceptable for 10 CFR Part 50, Appendix K. Therefore, the change to TS 3.2.1 is acceptable.

The licensee also proposes deleting the definition of Average Planar Exposer from the TS Section 1.0 "Definitions." This change would allow the most suitable method to be utilized as specified in the COLR and will establish consistency in TS wording. This item is also removed from the table of contents. These changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 59588). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also relates to changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. Chatterton

Date: March 16, 1999