

April 13, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M89631 AND M89632)

Dear Mr. Farrar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 103 to Facility Operating License No. NPF-11 and Amendment No. 88 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 June 9, 1994.

The amendments implement a partial application of the Generic Electric ARTS (Average Power Range Monitor (APRM)/Rod Block Monitor (RBM)/Technical Specification) Improvement Program. Four new ARTS thermal limits replace the existing flow-referenced APRM trip setpoint setdown requirements and the Minimum Critical Power Ratio (MCPR) K_f factor. The extended load line limit analysis (ELLLA) portion of ARTS was previously implemented by Amendments 58 and 41 for Units 1 and 2, respectively. No changes to the Rod Block Monitor (RBM) will be made with this amendment.

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,

Original signed by:
William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

- Enclosures: 1. Amendment No. 103 to NPF-11
- 2. Amendment No. 88 to NPF-18
- 3. Safety Evaluation

cc w/encls: see next page

*See previous concurrence

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D. L. Farrar
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LaSalle County Station
Unit Nos. 1 and 2

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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. 103
License No. NPF-11

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the Commonwealth Edison Company (the licensee) dated June 9, 1994, 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 paragraphs 2.C.(2) and 2.C.(34)* of Facility Operating License No. NPF-11 are hereby amended to read as follows:

*Page 16a is attached, for convenience, for the composite license to reflect this change.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 103, 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.

(34) Deleted.

3. This amendment is effective upon date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: 1. License page 16a
2. Changes to the Technical Specifications

Date of Issuance: April 13, 1995

2.C.(34) Deleted.

ATTACHMENT TO LICENSE AMENDMENT NO. 103

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

IV	IV
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B 2-10	B 2-10
3/4 2-2	3/4 2-2
3/4 3-8	3/4 3-8
3/4 3-53a	3/4 3-53a
3/4 4-1	3/4 4-1
B 3/4 2-1	B 3/4 2-1
B 3/4 2-2	B 3/4 2-2
B 3/4 2-5	B 3/4 2-5
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LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-High 118% setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

3/4.2.2 INTENTIONALLY LEFT BLANK.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Within 24 hours prior to startup, if not performed within the previous 7 days.
- (d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER \geq 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF $>$ 1.02. In addition, adjust any APRM channel within 12 hours, if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is $<$ 0.98. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

- (e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) Measure and compare core flow to rated core flow.
- (h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
5. <u>SCRAM DISCHARGE VOLUME</u>		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. <u>REACTOR COOLANT SYSTEM RECIRCULATION FLOW</u>		
a. Upscale	≤ 108/125 of full scale	≤ 111/125 of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W).

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

- a. With only one (1) reactor coolant system recirculation loop in operation, comply with Specification 3.4.1.5 and:
 1. Within four (4) hours:
 - a) Place the recirculation flow control system in the Master Manual mode or lower, and
 - b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and
 - c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,
 - d) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6.
 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.
- b. With no reactor coolant recirculation loops in operation:
 1. Take the ACTION required by Specification 3.4.1.5, and
 2. Be in at least HOT SHUTDOWN within the next six (6) hours.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor.

However, the current General Electric (GE) calculational models (SAFER/GESTR described in Reference 3), which are consistent with the requirements of Appendix K to 10 CFR 50, have established that APLHGR values are not expected to be limited by LOCA/ECCS considerations. APLHGR limits are still required, however, to assure that fuel rod mechanical integrity is maintained. They are specified for all resident fuel types in the Core Operating Limit Report based on the fuel thermal-mechanical design analysis.

The purpose of the power- and flow-dependent MAPLHGR factors specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At less than 100% of rated flow or rated power, the required MAPLHGR is the minimum of either (a) the product of the rated MAPLHGR limit and the power-dependent MAPLHGR factor or (b) the product of the rated MAPLHGR limit and the flow-dependent MAPLHGR factor. The power- and flow-dependent MAPLHGR factors assure that the fuel remains within the fuel design basis during transients at off-rated conditions. Methodology for establishing these factors is described in Reference 6.

POWER DISTRIBUTION SYSTEMS

BASES

3/4.2.2 DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in the CORE OPERATING LIMITS REPORT.

Analyses have been performed to determine the effects on CRITICAL POWER RATIO (CPR) during a transient assuming that certain equipment is out of service. A detailed description of the analyses is provided in Reference 5. The analyses performed assumed a single failure only and established the licensing bases to allow continuous plant operation with the analyzed equipment out of service. The following single equipment failures are included as part of the transient analyses input assumptions:

- 1) main turbine bypass system out of service,
- 2) recirculation pump trip system out of service,

POWER DISTRIBUTION SYSTEMS

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

The value for τ_b used in Specification 3.2.3 is 0.687 seconds which is conservative for the following reason:

For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^n N_i$ of 598 was used. This represents one full core data set at BOC plus one full core data set following a 120 day outage plus twelve 10% of core, 19 rods, data sets. The 12 data sets are equivalent to 24 operating months of surveillance at the increased surveillance frequency of one set per 60 days required by the action statements of Specifications 3.1.3.2 and 3.1.3.4.

That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

The purpose of the power- and flow-dependent MCPR limits specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR assures that the Safety Limit MCPR will not be violated. Methodology for establishing the power- and flow-dependent MCPR limits is described in Reference 6.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

POWER DISTRIBUTION SYSTEMS

BASES

3/4.2.4 LINEAR HEAT GENERATION RATE

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 power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566A, September 1986.
2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Company Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980, from R. H. Buchholz (GE) to P. S. Check (NRC).
3. "LaSalle County Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," General Electric Company Report NEDC-32258P, October 1993.
4. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).
5. "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Station Units 1 and 2," NEDC-31455, November 1987.
6. "ARTS Improvement Program Analysis for LaSalle County Units 1 and 2," General Electric Company Report NEDC-31531P, December 1993.

INSTRUMENTATION

BASES

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specification in Section 3/4.1.4, Control Rod Program Controls. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

ADMINISTRATIVE CONTROLS

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 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 (τ), dependent MCPR limits, and power and flow dependent MCPR limits) 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).



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. 88
License No. NPF-18

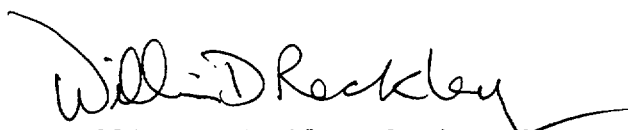
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment filed by the Commonwealth Edison Company (the licensee) dated June 9, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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. 88, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective upon date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



William D. Reckley, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 88

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

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

REMOVE

IV
XII
B 2-10
3/4 2-2
3/4 3-8
3/4 3-54
3/4 4-1
B 3/4 2-1
B 3/4 2-2
B 3/4 2-5
B 3/4 2-6
B 3/4 3-4
6-25

INSERT

IV
XII
B 2-10
3/4 2-2
3/4 3-8
3/4 3-54
3/4 4-1
B 3/4 2-1
B 3/4 2-2
B 3/4 2-5
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LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-High 118% setpoint; i.e., for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

3/4.2.2 INTENTIONALLY LEFT BLANK.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL FUNCTIONAL TEST</u>	<u>CHANNEL CALIBRATION</u>	<u>OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED</u>
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

- (a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
- (b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
- (c) Within 24 hours prior to startup, if not performed within the previous 7 days.
- (d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER \geq 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF $>$ 1.02. In addition, adjust any APRM channel within 12 hours, if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is $<$ 0.98. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

- (e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
- (f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
- (g) Measure and compare core flow to rated core flow.
- (h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

<u>TRIP FUNCTION</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUE</u>
5. <u>SCRAM DISCHARGE VOLUME</u>		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. <u>REACTOR COOLANT SYSTEM RECIRCULATION FLOW</u>		
a. Upscale	≤ 108/125 of full scale	≤ 111/125 of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W).

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

- a. With only one (1) reactor coolant system recirculation loop in operation, comply with Specification 3.4.1.5 and:
 1. Within four (4) hours:
 - a) Place the recirculation flow control system in the Master Manual mode or lower, and
 - b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2, and
 - c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation by 0.01 per Specification 3.2.3, and,
 - d) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6.
 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.
- b. With no reactor coolant recirculation loops in operation:
 1. Take the ACTION required by Specification 3.4.1.5, and
 2. Be in at least HOT SHUTDOWN within the next six (6) hours.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. This specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure dependent steady-state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor.

However, the current General Electric (GE) calculational models (SAFER/GESTR described in Reference 3), which are consistent with the requirements of Appendix K to 10 CFR 50, have established that APLHGR values are not expected to be limited by LOCA/ECCS considerations. APLHGR limits are still required, however, to assure that fuel rod mechanical integrity is maintained. They are specified for all resident fuel types in the Core Operating Limit Report based on the fuel thermal-mechanical design analysis.

The purpose of the power- and flow-dependent MAPLHGR factors specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At less than 100% of rated flow or rated power, the required MAPLHGR is the minimum of either (a) the product of the rated MAPLHGR limit and the power-dependent MAPLHGR factor or (b) the product of the rated MAPLHGR limit and the flow-dependent MAPLHGR factor. The power- and flow-dependent MAPLHGR factors assure that the fuel remains within the fuel design basis during transients at off-rated conditions. Methodology for establishing these factors is described in Reference 6.

POWER DISTRIBUTION SYSTEMS

BASES

3/4.2.2 DELETED

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and presented in the CORE OPERATING LIMITS REPORT.

Analyses have been performed to determine the effects on CRITICAL POWER RATIO (CPR) during a transient assuming that certain equipment is out of service. A detailed description of the analyses is provided in Reference 5. The analyses performed assumed a single failure only and established the licensing bases to allow continuous plant operation with the analyzed equipment out of service. The following single equipment failures are included are part of the transient analyses input assumptions:

1. main turbine bypass system out of service,
2. recirculation pump trip system out of service,

POWER DISTRIBUTION SYSTEMS

BASES

MINIMUM CRITICAL POWER RATIO (Continued)

The value for τ_B used in Specification 3.2.3 is 0.687 seconds which is conservative for the following reason:

For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^n N_i$ of 598 was used. This represents one full core data set at BOC plus one full core data set following a 120 day outage plus twelve 10% of core, 19 rods, data sets. The 12 data sets are equivalent to 24 operating months of surveillance at the increased surveillance frequency of one set per 60 days required by the action statements of Specifications 3.1.3.2 and 3.1.3.4.

That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

The purpose of the power- and flow-dependent MCPR limits specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow and core power conditions. At a given power and flow operating condition, the required MCPR is the maximum of either the power-dependent MCPR limit or the flow-dependent MCPR limit. The required MCPR assures that the Safety Limit MCPR will not be violated. Methodology for establishing the power- and flow-dependent MCPR limits is described in Reference 6.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

POWER DISTRIBUTION SYSTEMS

BASES

3/4.2.4 LINEAR HEAT GENERATION RATE

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 power spike penalty specified is based on the analysis presented in Section 3.2.1 of the GE topical report NEDM-10735 Supplement 6, and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design LINEAR HEAT GENERATION RATE due to power spiking.

References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDO-20566A, September 1986.
2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980, from R. H. Buchholz (GE) to P. S. Check (NRC).
3. "LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis," General Electric Co. Report NEDC-32258P, October 1993.
4. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).
5. "Extended Operating Domain and Equipment Out-of-Service for LaSalle County Nuclear Station Units 1 and 2," NEDC-31455, November 1987.
6. "ARTS Improvement Program Analysis for LaSalle County Station Units 1 and 2," General Electric Co. Report NEDC-31531P, December 1993.

INSTRUMENTATION

BASES

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specification in Section 3/4.1.4, Control Rod Program Controls. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes," April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

Core Operating Limits 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 power and flow dependent MCPR limits) 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.

B. Deleted.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 103 TO FACILITY OPERATING LICENSE NO. NPF-11 AND
AMENDMENT NO. 88 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 June 9, 1994, Commonwealth Edison Company (ComEd, the licensee) requested changes to the LaSalle County Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). The proposed changes involve implementing the remaining thermal limits portion of the General Electric APRM/RBM/Technical Specification (ARTS) Improvement Program. The licensee proposed that four new ARTS thermal limits replace the existing flow-referenced average power range monitor (APRM) trip setpoint setdown requirements and the minimum critical power ratio (MCPR) K_f factor. The extended load line limit analysis (ELLLA) portion of ARTS was previously implemented by Amendments 58 and 41 for Units 1 and 2, respectively. No changes to the rod block monitor (RBM) will be made with this amendment. In support of the request, the licensee submitted (1) a description and evaluation of the changes, (2) the proposed new TS, (3) identification of the new thermal limits in the Core Operating Limits Report (COLR), and (4) a General Electric (GE) analysis of the ARTS Improvement Program for LaSalle (Ref. 2).

The licensee stated that the implementation of the thermal limits of the GE ARTS Improvement Program would result in increased operational flexibility at LaSalle by incorporating the fuel peaking limits in the power and flow dependent thermal limits. Additionally, the licensee requested the removal of the Unit 1 Facility Operating License section which refers to the T-factor for the first fuel cycle of operation and is, therefore, no longer applicable.

2.0 EVALUATION

LaSalle's current Technical Specifications (TS) require that the flow-referenced APRM trips be lowered or the APRM readings increased by up to 10% when the maximum fraction of limiting power density exceeds the fraction of rated power (T-factor correction). This requirement is associated with a now obsolete Hench-Levy Minimum Critical Heat Flux Ratio criterion. GE has since developed the General Electric Thermal Analysis Basis (GETAB/GEXL) critical power correlation, which relies on boiling length and exit quality and

provides an acceptable alternative to the setdown requirement. The ARTS thermal limits ensure that the safety limit MCPR and fuel thermal-mechanical design bases are not violated during postulated transient events initiated from off-rated power and flow conditions. Changes resulting from the ARTS Improvement Program include:

- a. Elimination of the use of and reference to the MCPR K_f flow multiplication factor.
- b. Introduction of power and flow dependent adjustment factors for the maximum average planar linear heat generation rate (MAPLHGR) and MCPR limits.
- c. Revision of the COLR documentation requirements to include parameters used to determine the power and flow dependent MCPR and MAPLHGR limits for each cycle.
- d. Removal of the fraction of rated power and the maximum fraction of limiting power density requirements since they are used only in the determination of the required setdown of the APRM setpoints.

The proposed ARTS changes for LaSalle are common for GE boiling water reactors (BWRs). These options have been approved for several BWRs such as Hatch and Monticello in 1984, and Fermi 2, Pilgrim, and Cooper in 1991. The methodologies used for the safety analyses justifying the changes and establishment of new operating limits have been previously reviewed and approved by the staff. The proposed APRM changes are similar to changes approved for other reactors.

2.1 LOCA and Anticipated Operational Occurrences (AOOs) Analyses

General Electric evaluated the SAFER/GESTR LOCA analysis for LaSalle and concluded that the LOCA analysis results are not affected by the introduction of ARTS. The licensee stated that the current licensing basis peak cladding temperature of 1260 degrees will not change due to the application of the ARTS power and flow dependent limits. The staff has reviewed comparable GE examinations of the effect of ARTS on the SAFER/GESTR LOCA analysis and has concluded that suitable analyses were performed with similar results. Therefore, the staff agrees that the LaSalle LOCA analysis results should not be affected by the introduction of ARTS at LaSalle.

The LaSalle ARTS program uses transient analyses to define operating limits which conservatively assure that all licensing criteria are satisfied without setdown of the flow-referenced APRM scram setpoint. Specific LaSalle analyses were performed to confirm the applicability of generic power and flow dependent MCPR and MAPLHGR limits taken from the GE ARTS data base. These analyses included limiting transients such as feedwater controller failure (FWCF), turbine trip with no bypass (TTNBP), and load rejection with no bypass (LRNBP) for different power and flow conditions in order to develop plant specific MCPR and MAPLHGR limits. Description of the new limits is discussed

in Section 2.2. The plant limits were selected to remain valid through future reloads of GE fuels through GE9 utilizing the GEXL-PLUS correlation and the GEMINI analysis methods. The licensee stated that the GE analyses results showed that the FWCF event at 108% flow and off-rated conditions was more limiting than all other transients for power-dependent MCPR. Furthermore, the LRNBP and FWCF transients were analyzed assuming a combination of all currently allowed equipment out-of-service (EOMS). The staff has reviewed the ARTS transient analyses for LaSalle. Overall, the ARTS analyses and improvements to the APRM system parallel ARTS submittals for other BWRs which were accepted by the staff and, therefore, its use in the present amendment request is acceptable to the staff.

2.2 ARTS Thermal Limits

The GE analysis established four new limits which will be placed in the COLRs of both units. The four new limits include:

- a. A power-dependent MCPR limit, $MCPR_p$, which is a cycle-independent limit below 30% power, and a cycle-dependent limit above 30% power.
- b. A power-dependent MAPLHGR factor, $MAPFAC_p$.
- c. Flow-dependent MCPR limits, $MCPR_f$, for both automatic and manual flow control, which is cycle-independent for manual flow control, and cycle-dependent for automatic flow control, and replaced the current MCPR K_f multipliers for manual and automatic flow control.
- d. A flow-dependent MAPLHGR factor, $MAPFAC_f$.

Below 25 percent of full power, no MCPR and MAPLHGR limits are defined. However, the value for operation at 25% power, as shown on the appropriate plant specific MCPR and MAPLHGR curves, may be used when operating below 25% power. In the interval between 25 and 30 percent of full power, flow-dependent effects are taken into account by having two power-dependent curves (one for operation at greater than 50% flow and one for operation at equal to or less than 50% flows). For this region, the values shown on the two flow-dependent curves are the total power-dependent $MCPR_p$ and $MAPFAC_p$ limits and are not cycle-dependent. Analyses of the generation of the power-dependent $MCPR_p$ and $MAPFAC_p$ curves is presented in NEDC-31531P, "ARTS Improvement Program Analysis for LaSalle Units 1 and 2," General Electric Nuclear Energy, December 1993.

When power is greater than or equal to 30 percent of full power, the $MCPR_p$ is the product of the full power cycle-specific operating limits maximum critical power ratio (OLMCPR) and the power-dependent, cycle-independent MCPR multiplier (K_p). The formulas for K_p are presented in NEDC-31531P. The power-dependent $MAPLHGR_p$ limit is the product of the fuel-type dependent MAPLHGR limit and the $MAPFAC_p$ value. The formulas for $MAPFAC_p$ are also presented in NEDC-31531P.

There are also flow-dependent MCPR and MAPLHGR values from 30% to 100% of rated core flow. These protect the safety limit MCPR and the fuel thermal-mechanical design bases during flow runout transients not terminated by a reactor scram. The flow-dependent MCPR limits ($MCPR_F$) for manual flow control replace the current MCPR K_f multiplier although $MCPR_F$ is not a multiplier. For manual flow control, the 102.5% line will be used. The maximum flow for the slow flow runout event is 102.5% and is due to a single failure which causes one recirculation loop to ramp to its maximum position. For operation in automatic flow control, the 117% maximum flow $MCPR_F$ curve for manual flow control is adjusted so that the limit intercepts 100% flow at the cycle-specific OLMCPR value. The flow-dependent MAPLHGR_F limit is the product of the fuel-type dependent MAPLHGR limit and the MAPFAC_F value. The formula for the MAPFAC_F values is presented in NEDC-31531P.

At a given power and flow condition, all four of the limits will be determined, and the most limiting MCPR and MAPLHGR will be chosen to be the governing limits by the unit's core monitoring software. The development of the adjustment factors described above supports both the implementation of the updated operating limits and the elimination of the APRM trip setpoint setdown requirements. Based on the above discussions, the staff concluded that the implementation of the ARTS Improvement Program at LaSalle is acceptable.

2.3 Technical Specification Changes for ARTS

The proposed changes to the LaSalle Technical Specifications (TS) are identified in the licensee's submittal. The changes include (1) deletion of the current setdown requirement, (2) new power- and flow-dependent MCPR and MAPLHGR limits, and (3) deletion of Unit 1 first fuel cycle section of the Facility Operating License which refers to the T-factor correlation. The proposed TS changes for Units 1 and 2 are as follows:

- (1) Index, P. IV, Section 3/4.2.2, "APRM Setpoints," deleted as noted.
- (2) Index, P. XII, Section 3/4.2.2, "APRM Setpoints," deleted as noted.
- (3) B 2.2.1.2, "Average Power Range Monitor," delete the reference made to Specification 3.2.2.
- (4) TS 3/4.2.2, "APRM Setpoints," deleted; T-factor is no longer applicable with the ARTS analysis.
- (5) Table 4.3.1.1, "Reactor Protection System Instrumentation Surveillance Requirements," deleted setdown requirement in Footnote (d).
- (6) Table 3.3.6-2, "Control Rod Withdrawal Block Instrumentation Setpoints," deleted reference to Specification 3.2.2 in Footnote (*).
- (7) TS 3.4.1.1, "Recirculation System," deleted reference to Specification 3.2.2 in Action a.1.d.

- (8) B 3/4.2.1, "Average Planar Linear Heat Generation Rate," added description of ARTS MAPLHGR limits.
- (9) B 3/4.2.2, "APRM Setpoints," deleted; TS 3/4.2.2 is deleted, therefore, Bases are no longer applicable.
- (10) B 3/4.2.3, "Minimum Critical Power Ratio," Added description of ARTS MCPR limits, deleted reference to K_f factor, and added ARTS analysis to reference list.
- (11) B 3/4.3.6, "Control Rod Withdrawal Block Instrumentation," deleted reference to Section 3/4.2 since T-factor is deleted.
- (12) 6.6.A.6.a.2, "Core Operating Limits Report," modified explanation of MCPR.

Unit specific TS changes included:

Unit 1

- (1) FOL, 2.C.(34), deleted - no longer applicable; this applied to Unit 1 Cycle 1 only.

Unit 2

- (1) 6.6.A.6.a.2, editorial change to the title from "Semiannual Radioactive Effluent Release Report" to correct name of "Core Operating Limits Report."

The changes implement the thermal limits portion of the GE ARTS Improvement Program. The analyses presented examined the same areas as previous ARTS submittals reviewed by the staff. The methods used have been previously approved and the results of the analyses fall within accepted limits. The staff concludes that the results submitted by the licensee justify the proposed ARTS changes to LaSalle Units 1 and 2.

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 requirements 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 (59 FR 39582). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

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