ATTACHMENT B

PROPOSED CHANGES TO THE LICENSE/TECHNICAL SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

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(d) The maximum average planar linear heat generation (MAPLHGR) limit will be reduced by 0.85.

(e) Technical Specification Setpoints shall read as follows:

T.S.2.2.1 S 0.66W + 45.7 (Trip Setpoint)
S 0.66W + 48.7 (Allowable)

T.S.3.2.2 S (0.66W + 45.7) T*
S_{RB} (0.66W + 36.7) T*
T* as defined in T.S.3.2.2

T.S.3.3.6 APRM Upscale 0.66W + 36.7 (Trip Setpoint)
APRM Upscale 0.66W + 39.7 (Allowable)
RBM Upscale 0.66W + 34.7 (Trip Setpoint)
RBM Upscale 0.66W + 37.7 (Allowable)

(f) The average power range monitor (APRM) flux noise will be measured once per shift; and the recirculation loop flow will be reduced if the flux noise averaged over 1/2 hour exceeds 5 percent peak to peak, as measured by the APRM chart recorder.

(g) The core plate delta P noise will be measured once per shift, and the recirculation loop flow will be reduced if the noise exceeds one (1) psi peak-to-peak.

Am. 12
12/20/82

D. Exemptions from certain requirements of Appendices G, H and J and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2 and No. 3 to the Safety Evaluation Report. In addition, an exemption was requested until the completion of the first refueling from the requirements of 10 CFR 70.24 and an exemption from 10 CFR Part 50, Appendix E from performing a full scale exercise within one year before issuance of an operating license, both exemptions are described in Supplement No. 2 of the Safety Evaluation Report. Finally, an exemption was requested from the requirements of 10 CFR 50.44 until either the required 100 percent rated thermal power trip startup test has been completed or the reactor has operated for 120 effective full power days as specified by the Technical Specifications. This latter exemption is described in the safety evaluation of License Amendment No. 12. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

(iii) Prior to exceeding five percent power, the licensee shall include a description of the dose calculational methodology with a Class A transport and diffusion module, and a description of an acceptable meteorological measurement preventative and corrective maintainance program in the radiological emergency plan.

Am. 3 7/15/82 (31) Bolting of Valves

Prior to January 15, 1983, the licensee shall check the torque on all non-pressure boundary bolts (bolts whose failure will effect the operability of the valve) on each safety-related valve located outside containment.

Am. 4 8/13/82 (32) Vacuum Breaker Valves

Prior to November 1, 1982, the licensee shall complete a test and shall submit its evaluation of the results which confirm the capability of the vacuum breaker valves to withstand the opening and closing forces associated with pool swell.

Am. 4 8/13/82 (33) Heating-Ventilation and Air Condition Systems

(a) Prior to exceeding five percent power operation, the licensee must provide formal documentation of information regarding HVAC design fabrication and installation, discussed in meetings with the NRC on August 2 and 4, 1982.

(b) Prior to exceeding fifty percent power operation, the licensee shall submit the results of an independent review acceptable to the NRC staff of the HVAC system, including design changes, fabrication, and installation. The review shall encompass all safety- related HVAC systems and the effect of non-safety related HVAC system failures on safety systems.

e

Am. 11 12/16/82 (34) Through the First Fuel Cycle of Plant Operation, Technical Specification 3.4.1.1 is Modified for One Recirculation Loop out of Service with Provisions

(a) The steady-state thermal power level will not exceed 50 percent of rated power.

(b) The minimum critical power ratio (MCPR) safety limit will be increased by 0.01 to 1.07.

(c) The minimum critical power ratio limiting condition for operation (LCO) will be increased by 0.01.

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP.

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.

PAGE 3/4 2-2

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3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased simulated thermal power-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

a. Two Recirculation Loop Operation
S less than or equal to (0.58W + 59%)T
S_{RB} less than or equal to (0.58W + 47%)T

b. Single Recirculation Loop Operation
S less than or equal to (0.58W + 54.3%)T
S_{RB} less than or equal to (0.58W + 42.3%)T

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 108.5 million lbs/hr,
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY or the value 1.0. T is always less than or equal to 1.0.

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

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased simulated thermal power-upscale control rod block trip setpoint set less conservatively than S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} 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

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and control rod block trip setpoint verified to be within the above limits or adjusted, as required:

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 MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
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 ≥ 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, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. 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.

No Changes
INFO Only

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
The Rod Block Monitor Upscale Setpoints shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.		
b. Inoperative	N.A.	N.A.
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	$\leq$ 0.58 W + 47%*	$\leq$ 0.58 W + 50%*
2) Single Recirculation Loop Operation	$\leq$ 0.58 W + 42.3%	$\leq$ 0.58 W + 45.3%*
b. Inoperative	N.A.	N.A.
c. Downscale	$\geq$ 5% of RATED THERMAL POWER	$\geq$ 3% of RATED THERMAL POWER
d. Neutron Flux-High	$\leq$ 12% of RATED THERMAL POWER	$\leq$ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$\leq 2 \times 10^5$ cps	$\leq 5 \times 10^5$ cps
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 0.7 cps	$\geq$ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	$\leq$ 108/125 of full scale	$\leq$ 110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	$\geq$ 5/125 of full scale	$\geq$ 3/125 of full scale

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
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). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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, 3.2.2, 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.

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.

<< Insert Paragraph #1 >>

Paragraph #1

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.

DELETED

3/4.2.2 ~~APRM SETPOINTS~~

~~The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that ≥ 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.~~

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,

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 K_f factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the K_f factor. The K_f factor assures that the Safety Limit MCPR will not be violated. Methodology for establishing the K_f factor is described in Reference 4.

<< Insert Paragraph #2 >>

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.

Paragraph #2

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.

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 Co. Report NEDC-31531P, December 1993

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 specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. 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.

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% a scram time, tau (t), dependent MCPR limits, and ~~K_f core flow~~ power and flow dependent MCPR ~~adjustment factors~~ 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).

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

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. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP.

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.

PAGE 3/4 2-2

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~~3/4 2.2 APRM SETPOINTS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased simulated thermal power-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:~~

~~a. Two Recirculation Loop Operation~~

~~S less than or equal to (0.58W + 59%)T~~

~~S_{RB} less than or equal to (0.58W + 47%)T~~

~~b. Single Recirculation Loop Operation~~

~~S less than or equal to (0.58W + 54.3%)T~~

~~S_{RB} less than or equal to (0.58W + 42.3%)T~~

~~where: S and S_{RB} are in percent of RATED THERMAL POWER,~~

~~W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 108.5 million lbs/hr,~~

~~T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY or the value 1.0. T is always less than or equal to 1.~~

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

~~ACTION:~~

~~With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased simulated thermal power-upscale control rod block trip setpoint set less conservatively than S or S_{RB}, as above determined, initiate corrective action within 15 minutes and restore S and/or S_{RB} 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~~

~~4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and control rod block trip setpoint verified to be within the above limits or adjusted, as required:~~

~~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 MFLPD greater than or equal to FRTP.~~

~~*With MFLPD greater than the FRTP up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.~~

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
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 ≥ 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, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. 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.

No Changes
INFO Only

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
The Rod Block Monitor Upscale Setpoints shall be established according to the relationships specified in the CORE OPERATING LIMITS REPORT.		
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power-Upscale		
1) Two Recirculation Loop Operation	≤ 0.58 W + 47%*	≤ 0.58 W + 50%*
2) Single Recirculation Loop Operation	< 0.58 W + 42.3%	< 0.58 W + 45.3%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux-High	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	< 2×10^5 cps	< 5×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	< 108/125 of full scale	< 110/125 of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale

TABLE 3.3.6-2 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 765' 5¼"	≤ 765' 5¼"
b. Scram Discharge Volume Switch in Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
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). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

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, 3.2.2, 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.

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.

<< Insert Paragraph #1 >>

Paragraph #1

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.

DELETED

3/4.2.2 ~~APRM SETPOINTS~~

~~The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.~~

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,

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 K_f factor specified in the CORE OPERATING LIMITS REPORT is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the K_f factor. The K_f factor assures that the Safety Limit MCPR will not be violated. Methodology for establishing the K_f factor is described in Reference 4.

<< Insert Paragraph #2 >>

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.

Paragraph #2

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.

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 Station Units 1 and 2," General Electric Co. Rept NEDC-31531P, December 1993

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 specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. 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.

~~Semiannual Radioactive Effluent Release~~ 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 ~~Kf core flow~~ Power and flow dependent MCPR ~~adjustment factors~~ 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.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

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

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

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

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

<u>The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:</u>

The probability of an accident previously evaluated will not increase as a result of this change, because no changes to plant systems will occur. All changes are related to core monitoring software, and there will be no physical changes to equipment.

The consequences of an accident previously evaluated will not increase as a result of the proposed changes. The power- and flow-dependent MCPR and MAPLHGR limits incorporate sufficient conservatism so the safety limit MCPR (operating limit MCPR for automatic flow control) and the fuel thermal-mechanical limits will not be violated for any power and flow condition. Because these limits are protected during normal operation, the consequences of any transient will not increase with this change in limit definition. General Electric has verified in Attachment E that the introduction of Arts will not cause any change in the Licensing Basis PCT resulting from a Loss-Of-Coolant Accident, nor any change in the results satisfying the other LOCA acceptance criteria of 10CFR 50.46 and Section 15.6.5 of NUREG-0800 (Standard Review Plan), which are: cladding oxidation, metal-water reaction (hydrogen generation), coolable geometry and long-term cooling.

<u>The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because:</u>

Since no physical changes to any plant system are occurring, there will be no new or different types of accidents created by this change. No interactions between equipment systems will be changed in any manner.

<u>The proposed changes do not involve a significant reduction in a margin of safety because:</u>

The power- and flow-dependent MCPR and MAPLHGR limits will sufficiently protect the SLMCPR (OLMCPR for automatic flow control) and the fuel thermal-mechanical limits at all power and flow conditions. The ARTS limits conservatively assure that all licensing criteria are satisfied without setdown of the flow referenced APRM scram and rod block trips. The limits were developed using NRC approved methods, and satisfy the same NRC approved criteria that the APRM setdown requirement does.

ATTACHMENT C

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

CONCLUSION

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are not considered likely to involve significant hazards considerations. This proposed amendment most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan, section 4.4, "Thermal and Hydraulic Design".

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92 (c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT E

GE ARTS ANALYSIS

FOR

LASALLE COUNTY STATION UNITS 1 AND 2

ATTACHMENT F

WITHOLDING AFFIDAVIT

FOR

GENERAL ELECTRIC ARTS ANALYSIS REPORT

General Electric Company

AFFIDAVIT

I, **David J. Robare,** being duly sworn, depose and state as follows:

(1) I am Project Manager, Plant Licensing/Renewal Projects, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-31531P, *ARTS Improvement Program Analysis for LaSalle County Station Units 1 and 2,* Class III (GE Company Proprietary Information), dated December 1993. The proprietary information is delineated by bars marked in the margin adjacent to the specific material.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in both paragraphs (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of plant improvements to

increase operational flexibility and efficiency for the BWR.

The development and approval of the BWR loss-of-coolant accident analysis computer codes used in this analysis was achieved at a significant cost, on the order of several million dollars, to GE.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

STATE OF CALIFORNIA)
) ss:
COUNTY OF SANTA CLARA)

David J. Robare, being duly sworn, deposes and says:

That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 14TH day of DECEMBER 1993.

David J. Robare
General Electric Company

Subscribed and sworn before me this 14th day of December 1993.

Mary L. Kendall
Notary Public, State of California

MARY L. KENDALL
COMM. # 987864
Notary Public — California
SANTA CLARA COUNTY
My Comm. Expires MAR 26, 1997

12/13/93RTH