

October 29, 1996

Ms. Irene M. Johnson, Acting 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. M95156 AND M95157)

Dear Ms. Johnson:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 116 to Facility Operating License No. NPF-11 and Amendment No. 101 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 April 8, 1996, as supplemented by letter dated October 14, 1996.

The amendments change various sections of the Technical Specifications (TS) to reflect the transition of fuel supplier from General Electric (GE) to Siemens Power Corporation (SPC). The amendments revise the definitions, Limiting Conditions for Operation, required actions, or surveillance requirements related to Linear Heat Generation Rate, Critical Power Ratio, Minimum Critical Power Ratio, and Average Planar Linear Heat Generation Rate to incorporate SPC methodology and to delete GE terminology. The amendments also include changes to Section 6.0 of the TS to include SPC references, relocate the requirements for the traversing in-core probe system from the TS to the Core Operating Limits Report, and revise the fuel description in TS Section 5.0.

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,

/s/

Donna M. Skay, Project Manager  
Project Directorate III-2  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures: 1. Amendment No. 116 to NPF-11  
2. Amendment No. 101 to NPF-18  
3. Safety Evaluation

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

October 29, 1996

Ms. Irene M. Johnson, Acting Manager  
Nuclear Regulatory Services  
Commonwealth Edison Company  
Executive Towers West III  
1400 Opus Place, Suite 500  
Downers Grove, IL 60515

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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,

A handwritten signature in cursive script, appearing to read "Donna M. Skay".

Donna M. Skay, Project Manager  
Project Directorate III-2  
Division of Reactor Projects - III/IV  
Office of Nuclear Reactor Regulation

Docket Nos. 50-373, 50-374

Enclosures: 1. Amendment No. 116 to NPF-11  
2. Amendment No. 101 to NPF-18  
3. Safety Evaluation

cc w/encl: see next page

I. Johnson  
Commonwealth Edison Company

LaSalle County Station  
Unit Nos. 1 and 2

cc:

Phillip P. Steptoe, Esquire  
Sidley and Austin  
One First National Plaza  
Chicago, Illinois 60603

Robert Cushing  
Chief, Public Utilities Division  
Illinois Attorney General's Office  
100 West Randolph Street  
Chicago, Illinois 60601

Assistant Attorney General  
100 West Randolph Street  
Suite 12  
Chicago, Illinois 60601

Michael I. Miller, Esquire  
Sidley and Austin  
One First National Plaza  
Chicago, Illinois 60603

U.S. Nuclear Regulatory Commission  
Resident Inspectors Office LaSalle Station  
2605 N. 21st Road  
Marseilles, Illinois 61341-9756

Document Control Desk-Licensing  
Commonwealth Edison Company  
1400 Opus Place, Suite 400  
Downers Grove, Illinois 60515

Chairman  
LaSalle County Board of Supervisors  
LaSalle County Courthouse  
Ottawa, Illinois 61350

Attorney General  
500 South Second Street  
Springfield, Illinois 62701

Chairman  
Illinois Commerce Commission  
Leland Building  
527 East Capitol Avenue  
Springfield, Illinois 62706

Illinois Department of Nuclear Safety  
Office of Nuclear Facility Safety  
1035 Outer Park Drive  
Springfield, Illinois 62704

Regional Administrator  
U.S. NRC, Region III  
801 Warrenville Road  
Lisle, Illinois 60532-4351

LaSalle Station Manager  
LaSalle County Station  
Rural Route 1  
P.O. Box 220  
Marseilles, Illinois 61341



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. 116  
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 April 8, 1996, as supplemented on October 14, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;
  - C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

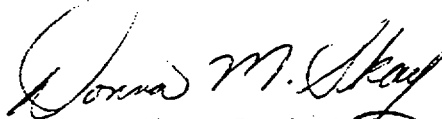
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(2) Technical Specifications and Environmental Protection Plan

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

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

FOR THE NUCLEAR REGULATORY COMMISSION



Donna M. Skay, 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: October 29, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. NPF-11

DOCKET NO. 50-373

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

REMOVE

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## 1.0 DEFINITIONS

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

### ACTION

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

### AVERAGE PLANAR EXPOSURE

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

### AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

### CHANNEL CALIBRATION

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

### CHANNEL CHECK

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

### CHANNEL FUNCTIONAL TEST

- 1.6 A CHANNEL FUNCTIONAL TEST shall be:
- a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.
  - b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

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

## DEFINITIONS

### CORE ALTERATION

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

### CORE OPERATING LIMITS REPORT

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

### CRITICAL POWER RATIO

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

### DOSE EQUIVALENT I-131

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

### $\bar{E}$ -AVERAGE DISINTEGRATION ENERGY

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

### EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

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

## DEFINITIONS

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

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

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

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

#### 1.14 DELETED

### FRACTION OF RATED THERMAL POWER

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

### FREQUENCY NOTATION

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

### GASEOUS RADWASTE TREATMENT SYSTEM

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

### IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

- a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

### ISOLATION SYSTEM RESPONSE TIME

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

## DEFINITIONS

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### 1.20 DELETED

#### LIMITING CONTROL ROD PATTERN

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

#### LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.

#### LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e, all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

### 1.24 DELETED

#### MEMBERS(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

#### MINIMUM CRITICAL POWER RATIO

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

#### OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

## 2.1 SAFETY LIMITS

### BASES

2.0 The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

#### 2.1.1 THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than  $0.10 \times 10^6$  lbs/hr-ft<sup>2</sup> (equivalent to a core flow of less than 10%) or pressures less than 590 psia. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of  $28 \times 10^3$  lbs/hr (approximately a mass velocity of  $0.25 \times 10^6$  lbs/hr-ft<sup>2</sup>), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than  $28 \times 10^3$  lbs/hr. Full-scale ATLAS test data taken at pressures from 14.7 to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

## SAFETY LIMITS

### BASES

#### 2.1.2 THERMAL POWER, High Pressure and High Flow

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

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

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-5. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.
2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.
3. ANFB Critical Power Correlation, ANF-1125(P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.



## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### BASES

#### 2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

##### 1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

##### 2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

## REACTIVITY CONTROL SYSTEM

### 3/4.1.2 REACTIVITY ANOMALIES

#### LIMITING CONDITION FOR OPERATION

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3.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

With the reactivity different by more than 1% delta k/k:

- a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.
- b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

#### SURVEILLANCE REQUIREMENTS

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4.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall be verified to be less than or equal to 1% delta k/k:

- a. During the first startup following CORE ALTERATIONS, and
- b. At least once per 31 effective full power days during POWER OPERATION.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

SURVEILLANCE REQUIREMENTS

4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

4.2.3.2 The applicable MCPR limit shall be determined from the COLR based on:

- a. Technical Specification Scram Speed (TSSS) MCPR limits, or
- b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR.

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS).

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### 3/4.4 REACTOR COOLANT SYSTEM

#### 3/4.4.1 RECIRCULATION SYSTEM

##### RECIRCULATION LOOPS

##### LIMITING CONDITION FOR OPERATION

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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 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.
    - e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.
  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

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least  $R + 0.38\% \Delta K$  or  $R + 0.28\% \Delta K$ , as appropriate. The value of  $R$  in units of  $\% \Delta K$  is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of  $R$  must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning-of-life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored core  $k_{eff}$  and the predicted core  $k_{eff}$  of  $1\% \Delta k/k$  has been established based on engineering judgment. Alternatively predicted control rod configuration can be compared with actual control rod configuration, and shown to be within  $1\% \Delta k/k$ . A  $> 1\%$  deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

BASES

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3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Volume 1 and Supplements 1 and 2, March 1983.

BASES

3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 50 to 125 minutes. A normal quantity of 4587 gallons net of solution having a 13.4% sodium pentaborate concentration is required to meet a shutdown requirement of 3%. There is an additional allowance of 25% in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel.

The temperature requirement on the sodium pentaborate solution is necessary to maintain the solubility of the solution as it was initially mixed to the appropriate concentration. Checking the volume of fluid and the temperature once each 24 hours assures that the solution is available for injection.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.



## 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

##### GE Fuel

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 8), 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, for GE fuel, to assure that fuel rod mechanical integrity is maintained. They are specified for all GE fuel types in the CORE OPERATING LIMITS 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 9.

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

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#### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Con't)

##### SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed the peak cladding temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of APPENDIX K to 10 CFR Part 50. The models are described in Reference 1.

The 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 not strongly influenced by the rod-to-rod power distribution within the assembly.

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits for two-loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, an APLHGR limit corresponding to the product of the two-loop limit and a reduction factor specified in the COLR can be conservatively used to ensure that the PCT for single-loop operation is bound by the PCT for two-loop operation.

## POWER DISTRIBUTION SYSTEMS

### BASES

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#### 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.

The purpose of the power- and flow-dependent MCPR limits ( $MCPR_p$  and  $MCPR_f$ , respectively) specified in the CORE OPERATING LIMITS REPORT (COLR) is to define operating limits dependent on core flow and core power. 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 limit assures that the Safety Limit MCPR will not be violated.

The flow dependent MCPR limits ( $MCPR_f$ ) are established to protect the core from inadvertent core flow increases. The core flow increase event used to establish the limits is a slow flow runout to maximum flow that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1, Item 2). A conservative flow control line is used to define several core power/flow state points at which the analyses are performed.  $MCPR_f$  limits are established to support both the automatic and manual modes of operation. In the automatic mode,  $MCPR_f$  limits are established to protect the operating limit MCPR. For the manual mode, the limits are set to protect against violation of the safety limit MCPR.

The power-dependent MCPR limits, ( $MCPR_p$ ), are established to protect the core from plant transients other than core flow increases, including pressurization and the localized control rod withdrawal error events.

Analyses have been performed to determine the effects of assuming various equipment out-of-service scenarios on the (CPR) during transient events. Scenarios were performed to allow continuous plant operation with these systems

## BASES

### 3/4.2.3 MINIMUM CRITICAL POWER RATIO (Con't)

out of service. Appropriate MCPR limits and/or penalties are included in the COLR for each of the equipment out-of-service scenarios identified in the COLR. In some cases, the reported limits or penalties are based on a cycle-independent analysis, while in other cases, analyses are performed on a cycle-specific basis.

References 2-6 describe the methodology and codes used to evaluate the potentially bounding non-LOCA transient events identified in Chapter 15 of the UFSAR.

MCPR limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits takes advantage of improved scram insertion rates, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.1.3.3. If the scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times, the appropriate NSS MCPR limits identified in the COLR are applied. If the scram insertion times do not meet the NSS insertion criteria, the TSSS MCPR limits are applied.

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.

BASES3/4.2.4 LINEAR HEAT GENERATION RATEGE Fuel

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

SPC Fuel

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

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

References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.
2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.
3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.

BASES

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3/4.2.4 LINEAR HEAT GENERATION RATE

References Con't:

4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2, Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.
5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.
6. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, Exxon Nuclear Company, February 1987.
7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
8. LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-32258P, General Electric Company, October 1993.
9. ARTS Improvement Program analysis for LaSalle County Station Units 1 and 2, NEDC-31531P, General Electric Company, December 1993.

# INSTRUMENTATION

## BASES

### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

Analyses were performed to support continued operation with one or both trip systems of the EOC-RPT inoperable. The analyses provide MINIMUM CRITICAL POWER RATIO (MCPR) values which must be used if the EOC-RPT system is inoperable. These MCPR limits are included in the COLR and ensure that adequate margin to the MCPR safety limit exists with the EOC-RPT function inoperable. Application of these limits are discussed further in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

## INSTRUMENTATION

### BASES

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#### MONITORING INSTRUMENTATION (Continued)

##### 3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

##### 3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

##### 3/4.3.7.7 DELETED

##### 3/4.3.7.8 DELETED

##### 3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

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

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



## DESIGN FEATURES

### 5.3 REACTOR CORE

#### FUEL ASSEMBLIES

5.3.1 The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide ( $UO_2$ ) as fuel material. The bundles may contain water rods or water boxes. Limited substitutions of Zircalloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

#### CONTROL ROD ASSEMBLIES

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

### 5.4 REACTOR COOLANT SYSTEM

#### DESIGN PRESSURE AND TEMPERATURE

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

- a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
- b. For a pressure of:
  1. 1250 psig on the suction side of the recirculation pumps.
  2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
  3. 1500 psig from the discharge shutoff valve to the jet pumps.
- c. For a temperature of 575°F.

#### VOLUME

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

### 5.5 METEOROLOGICAL TOWER LOCATION

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

## ADMINISTRATIVE CONTROLS

### Monthly Operating 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) scram time, dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included.
  - (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. For LaSalle County Station Unit 1, the topical reports are:
  - (1) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
  - (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.
  - (3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.
  - (4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

## ADMINISTRATIVE CONTROLS

### Core Operating Limits Report (Continued)

- (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.
- (6) Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
- (7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.
- (8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.
- (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.
- (10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.
- (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.
- (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.
- (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.
- (14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.
- (15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

## ADMINISTRATIVE CONTROLS

### Core Operating Limits Report (Continued)

- (16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.
- (17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
- (18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).
- (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).
- (20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).
- (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).
- (22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.



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. 101  
License No. NPF-18

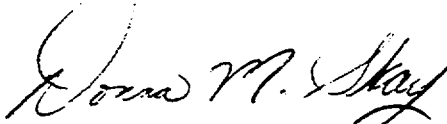
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment filed by the Commonwealth Edison Company (the licensee), dated April 8, 1996, as supplemented on October 14, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, 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. 101, 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 the date of issuance and shall be implemented prior to the startup of Cycle 8.

FOR THE NUCLEAR REGULATORY COMMISSION



Donna M. Skay, 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: October 29, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-18

DOCKET NO. 50-374

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

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## 1.0 DEFINITIONS

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

### ACTION

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

### AVERAGE PLANAR EXPOSURE

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

### AVERAGE PLANAR LINEAR HEAT GENERATION RATE

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

### CHANNEL CALIBRATION

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

### CHANNEL CHECK

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

### CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

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

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

## DEFINITIONS

### CORE ALTERATION

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

### CORE OPERATING LIMITS REPORT

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

### CRITICAL POWER RATIO

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

### DOSE EQUIVALENT I-131

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

### $\bar{E}$ -AVERAGE DISINTEGRATION ENERGY

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

### EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

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

### END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

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

## DEFINITIONS

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

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

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

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

1.14 DELETED

### FRACTION OF RATED THERMAL POWER

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

### FREQUENCY NOTATION

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

### GASEOUS RADWASTE TREATMENT SYSTEM

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

### IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

- a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

### ISOLATION SYSTEM RESPONSE TIME

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

1.20 DELETED

## DEFINITIONS

### LIMITING CONTROL ROD PATTERN

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

### LINEAR HEAT GENERATION RATE

1.22 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length. LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the CORE OPERATING LIMITS REPORT.

### LOGIC SYSTEM FUNCTIONAL TEST

1.23 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e, all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

1.24 Deleted

### MEMBERS(S) OF THE PUBLIC

1.25 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

### MINIMUM CRITICAL POWER RATIO

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

### OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

## 2.1 SAFETY LIMITS

### BASES

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

#### 2.1.1 THERMAL POWER, Low Pressure or Low Flow

For certain conditions of pressure and flow, the ANFB correlation is not valid for all critical power calculations. The ANFB correlation is not valid for bundle mass velocities less than  $0.10 \times 10^6$  lbs/hr-ft<sup>2</sup> (equivalent to a core flow of less than 10%) or pressures less than 590 psia. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of  $28 \times 10^3$  lbs/hr (approximately a mass velocity of  $0.25 \times 10^6$  lbs/hr-ft<sup>2</sup>), bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than  $28 \times 10^3$  lbs/hr. Full-scale ATLAS test data taken at pressures from 14.7 to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 Mwt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.



## SAFETY LIMITS

### BASES

#### 2.1.2 THERMAL POWER, High Pressure and High Flow

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

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

The bases for the uncertainties in system-related parameters are presented in NEDO-20340, Reference 2. The bases for the fuel-related uncertainties are found in References 1, 3-6. The uncertainties used in the analyses are provided in the cycle-specific transient analysis parameters document.

1. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524 (P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.
2. Process Computer Performance Evaluation Accuracy, NEDO-20340 and Amendment 1, General Electric Company, June 1974 and December 1974, respectively.
3. ANFB Critical Power Correlation, ANF-1125 (P)(A), and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
4. Advanced Nuclear Fuels Methodology for Boiling Water Reactors, XN-NF-80-19 (P)(A) Volume 1 Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
5. Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.
6. "Application of the ANFB Critical Power Correlation to Coresident GE Fuel for LaSalle Unit 2 Cycle 8," EMF-96-021(P), Revision 1, Siemens Power Corporation, February 1996; NRC SER letter dated September 26, 1996.

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## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### BASES

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#### 2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

##### 1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 21% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

##### 2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

## REACTIVITY CONTROL SYSTEM

### 3/4.1.2 REACTIVITY ANOMALIES

#### LIMITING CONDITION FOR OPERATION

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3.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall not exceed 1% delta k/k.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

With the reactivity different by more than 1% delta k/k:

- a. Within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.
- b. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

#### SURVEILLANCE REQUIREMENTS

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4.1.2 The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted critical control rod configuration shall be verified to be less than or equal to 1% delta k/k:

- a. During the first startup following CORE ALTERATIONS, and
- b. At least once per 31 effective full power days during POWER OPERATION.

## POWER DISTRIBUTION LIMITS

### 3/4.2.3 MINIMUM CRITICAL POWER RATIO

#### SURVEILLANCE REQUIREMENTS

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4.2.3.1 MCPR shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

4.2.3.2 The applicable MCPR limit shall be determined from the COLR based on:

- a. Technical Specification Scram Speed (TSSS) MCPR limits, or
- b. Nominal Scram Speed (NSS) MCPR limits if scram insertion times determined per surveillance 4.1.3.2 meet the NSS insertion times identified in the COLR.

Within 72 hours of completion of each set of scram testing, the results will be compared against the nominal scram speed (NSS) insertion times specified in the COLR, to verify the applicability of the transient analyses. Prior to initial scram time testing for an operating cycle, the MCPR operating limits used shall be based on the Technical Specification Scram Speeds (TSSS).

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### 3/4.4 REACTOR COOLANT SYSTEM

#### 3/4.4.1 RECIRCULATION SYSTEM

##### RECIRCULATION LOOPS

##### LIMITING CONDITION FOR OPERATION

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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 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.
    - e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.
  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.

## 3/4.1 REACTIVITY CONTROL SYSTEMS

### BASES

#### 3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least  $R + 0.38\% \Delta K$  or  $R + 0.28\% \Delta K$ , as appropriate. The value of  $R$  in units of  $\% \Delta K$  is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of  $R$  must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning-of-life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

#### 3/4.1.2 REACTIVITY ANOMALIES

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the Nuclear Design Methodology are larger than expected. A limit on the difference between the monitored core  $k_{eff}$  and the predicted core  $k_{eff}$  of  $1\% \Delta k/k$  has been established based on engineering judgment. Alternatively, predicted control rod configuration can be compared with actual control rod configuration, and shown to be within  $1\% \Delta k/k$ . A  $> 1\%$  deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.



## REACTIVITY CONTROL SYSTEMS

### BASES

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#### 3/4.1.3 CONTROL RODS (Continued)

In addition, the automatic CRD charging water header low pressure scram (see Table 2.2.1-1) initiates well before any accumulator loses its full capability to insert the control rod. With this added automatic scram feature, the surveillance of each individual accumulator check valve is no longer necessary to demonstrate adequate stored energy is available for normal scram action.

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod drive coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than 3.65 inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

#### 3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, Volume 1 and Supplements 1 and 2, March 1983."

## REACTIVITY CONTROL SYSTEMS

### BASES

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#### 3/4.1.4 CONTROL ROD PROGRAM CONTROLS (Continued)

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

#### 3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The standby liquid control system provides a backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. To meet this objective it is necessary to inject a quantity of boron which produces a concentration of 660 ppm in the reactor core in approximately 50 to 125 minutes. A normal quantity of 4587 gallons net of solution having a 13.4% sodium pentaborate concentration is required to meet a shutdown requirement of 3%. There is an additional allowance of 25% in the reactor core to account for imperfect mixing. The time requirement was selected to override the reactivity insertion rate due to cooldown following the Xenon poison peak and the required pumping rate is 41.2 gpm. The minimum storage volume of the solution is established to allow for the portion below the pump suction that cannot be inserted and the filling of other piping systems connected to the reactor vessel.

The temperature requirement on the sodium pentaborate solution is necessary to maintain the solubility of the solution as it was initially mixed to the appropriate concentration. Checking the volume of fluid and the temperature once each 24 hours assures that the solution is available for injection.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable.

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use.

Replacement of the explosive charges in the valves at regular intervals will assure that these valves will not fail because of deterioration of the charges.

## 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

##### GE Fuel

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 8), 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, for GE fuel, to assure that fuel rod mechanical integrity is maintained. They are specified for all GE fuel types in the CORE OPERATING LIMITS 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 9.

##### SPC Fuel

This specification assures that the peak cladding temperature of SPC fuel following a postulated design basis loss-of-coolant accident will not exceed

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

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#### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

the peak cladding temperature (PCT) and maximum oxidation limits specified in 10 CFR 50.46. The calculational procedure used to establish the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits is based on a loss-of-coolant accident analysis. The analysis is performed using calculational models which are consistent with the requirements of APPENDIX K to 10 CFR Part 50. The models are described in Reference 1.

The 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 not strongly influenced by the rod-to-rod power distribution within the assembly.

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limits for two-loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, an APLHGR limit corresponding to the product of the two-loop limit and a reduction factor specified in the COLR can be conservatively used to ensure that the PCT for single-loop operation is bound by the PCT for two-loop operation.

#### 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.

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

#### 3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

The purpose of the power- and flow-dependent MCPR limits ( $MCPR_p$  and  $MCPR_f$ , respectively) specified in the CORE OPERATING LIMITS REPORT (COLR) is to define operating limits dependent on core flow and core power. 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 limit assures that the Safety Limit MCPR will not be violated.

The flow dependent MCPR limits ( $MCPR_f$ ) are established to protect the core from inadvertent core flow increases. The core flow increase event used to establish the limits is a slow flow runout to maximum flow that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1, Item 2). A conservative flow control line is used to define several core power/flow state points at which the analyses are performed.  $MCPR_f$  limits are established to support both the automatic and manual modes of operation. In the automatic mode,  $MCPR_f$  limits are established to protect the operating limit MCPR. For the manual mode, the limits are set to protect against violation of the safety limit MCPR.

The power-dependent MCPR limits, ( $MCPR_p$ ), are established to protect the core from plant transients other than core flow increases, including pressurization and the localized control rod withdrawal error events.

Analyses have been performed to determine the effects of assuming various equipment out-of-service scenarios on the (CPR) during transient events. Scenarios were performed to allow continuous plant operation with these systems out of service. Appropriate MCPR limits and/or penalties are included in the COLR for each of the equipment out-of-service scenarios identified in the COLR. In some cases, the reported limits or penalties are based on a cycle-independent analysis, while in other cases, analyses are performed on a cycle-specific basis.

References 2-6 describe the methodology and codes used to evaluate the potentially bounding non-LOCA transient events identified in Chapter 15 of the UFSAR.

MCPR limits are presented in the CORE OPERATING LIMITS REPORT (COLR) for both Nominal Scram Speed (NSS) and Technical Specification Scram Speed (TSSS) insertion times. The negative reactivity insertion rate resulting from the scram plays a major role in providing the required protection against violating the Safety Limit MCPR during transient events. Faster scram insertion times provide greater protection and allow for improved MCPR performance. The application of NSS MCPR limits takes advantage of improved scram insertion rates, while the TSSS MCPR limits provide the necessary protection for the slowest allowable average scram insertion times identified in Specification 3.1.3.3. If the scram insertion times determined per

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

#### 3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

surveillance 4.1.3.2 meet the NSS insertion times, the appropriate NSS MCPR limits identified in the COLR are applied. If the scram insertion times do not meet the NSS insertion criteria, the TSSS MCPR limits are applied.

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.

#### 3/4.2.4 LINEAR HEAT GENERATION RATE

##### GE Fuel

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

##### SPC Fuel

The Linear Heat Generation Rate (LHGR) is a measure of the heat generation rate per unit length of a fuel rod in a fuel assembly at any axial location. LHGR limits are specified to ensure that fuel integrity limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). Operation above the LHGR limit followed by the occurrence of an AOO could potentially result in fuel damage and subsequent release of radioactive material. Sustained operation in excess of the LHGR limit could also result in exceeding the fuel design limits. The failure mechanism prevented by the LHGR limit that could cause fuel damage during AOOs is rupture of the fuel rod cladding caused by strain from the expansion of the fuel pellet. One percent plastic strain of the fuel cladding has been defined as the limit below which

## 3/4.2 POWER DISTRIBUTION LIMITS

### BASES

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#### 3/4.2.4 LINEAR HEAT GENERATION RATE (Continued)

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

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

#### References:

1. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.
2. Exxon Nuclear Methodology for Boiling Water Reactors, Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A), Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, March 1983.
3. Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3 Revision 2, Exxon Nuclear Company, January 1987.
4. Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A) Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.
5. COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.

### 3/4.2 POWER DISTRIBUTION LIMITS

#### BASES

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#### References: (Continued)

6. XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, Exxon Nuclear Company, February 1987.
7. Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
8. LaSalle County Station Units 1 and 2 SAFER/GESTR - LOCA Loss-of-Coolant Accident Analysis, NEDC-32258P, General Electric Company, October 1993.
9. ARTS Improvement Program analysis for LaSalle County Station Units 1 and 2, NEDC-31531P, General Electric Company, December 1993.



## INSTRUMENTATION

### BASES

#### 3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (continued)

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

#### 3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

Analyses were performed to support continued operation with one or both trip systems of the EOC-RPT inoperable. The analyses provide MINIMUM CRITICAL POWER RATIO (MCPR) values which must be used if the EOC-RPT system is inoperable. These MCPR limits are included in the COLR and ensure that adequate margin to the MCPR safety limit exists with the EOC-RPT function inoperable. Application of these limits are discussed further in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

## INSTRUMENTATION

### BASES

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#### MONITORING INSTRUMENTATION (Continued)

##### 3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

##### 3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions should not be made without this flux level information available to the operator. When the intermediate range monitors are on scale adequate information is available without the SRMs and they can be retracted.

##### 3/4.3.7.7 DELETED

##### 3/4.3.7.8 DELETED

##### 3/4.3.7.9 FIRE DETECTION INSTRUMENTATION

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

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

##### 3/4.3.7.10 DELETED

## DESIGN FEATURES

### 5.3 REACTOR CORE

#### FUEL ASSEMBLIES

5.3.1 The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide ( $UO_2$ ) as fuel material. The bundles may contain water rods or water boxes. Limited substitutions of Zircalloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

#### CONTROL ROD ASSEMBLIES

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

### 5.4 REACTOR COOLANT SYSTEM

#### DESIGN PRESSURE AND TEMPERATURE

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

- a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,
- b. For a pressure of:
  1. 1250 psig on the suction side of the recirculation pumps.
  2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.
  3. 1500 psig from the discharge shutoff valve to the jet pumps.
- c. For a temperature of 575°F.

#### VOLUME

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

### 5.5 METEOROLOGICAL TOWER LOCATION

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

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) scram time dependent MCPR limits, and power and flow dependent MCPR limits for Technical Specification 3.2.3. Effects of analyzed equipment out of service are included.
  - (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. For LaSalle County Station Unit 2, the topical reports are:
- (1) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
  - (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW<sup>TM</sup> Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.
  - (3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2 and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation November 1990.
  - (4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3, and 4, Advanced Nuclear Fuels Corporation, August 1990.
  - (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.
  - (6) Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F, and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.
  - (7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.
  - (8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.

Core Operating Limits Report (Continued)

- (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.
- (10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.
- (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.
- (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.
- (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.
- (14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.
- (15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.
- (16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.
- (17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
- (18) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).
- (19) Commonwealth Edison Topical Report NFSR-0085, "Benchmark of BWR Nuclear Design Methods," (latest approved revision).
- (20) Commonwealth Edison Topical Report NFSR-0085, Supplement 1, "Benchmark of BWR Nuclear Design Methods - Quad Cities Gamma Scan Comparisons," (latest approved revision).
- (21) Commonwealth Edison Topical Report NFSR-0085, Supplement 2, "Benchmark of BWR Nuclear Design Methods - Neutronic Licensing Analyses," (latest approved revision).
- (22) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

Core Operating Limits Report (Continued)

- 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

C. Unique Reporting Requirements

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

6.7 PROCESS CONTROL PROGRAM (PCP)\*

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

6.7.2 Licensee initiated changes to the PCP:

- a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.5.B.18. This documentation shall contain:
  - 1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and
  - 2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.
- b. Shall become effective upon review and acceptance by the Onsite Review and Investigative Function.

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



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-11 AND  
AMENDMENT NO. 101 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 April 8, 1996, as supplemented by letter dated October 14, 1996, Commonwealth Edison Company (ComEd, the licensee) proposed changes to the Technical Specifications to reflect the transition from General Electric (GE) fuel to Siemens Power Corporation (SPC) fuel. LaSalle County Station, Units 1 and 2 currently operate with GE fuel and methodologies. Beginning with Unit 2 Cycle 8 and Unit 1 Cycle 9, SPC fuel will be loaded with co-resident GE fuel. The ATRIUM-9B fuel design and SPC methodologies used to analyze fuel performance during normal and abnormal operating conditions were approved by the NRC. Technical Specification requirements related to the fuel thermal limits are affected as a result of the change in fuel design and methodologies. The current TS contain terminology and methodologies which are specific to GE fuel. Because both GE and SPC fuel will be present in the core, the licensee proposes to remove all vendor specific references from the TS requirements. The proposed TS will include discussions of both GE and SPC methodologies in the TS Bases. Other changes, unrelated to the transition to SPC fuel, were also proposed as line-item improvements from the Improved Standard Technical Specifications (NUREG-1434). These include relocation of the traversing in-core probe TS to the Core Operating Limits Reports (COLR) and revision of the fuel description in section 5.0.

The October 14, 1996, submittal provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Linear Heat Generation Rate (LHGR)

LHGR limits are monitored for GE fuel by the parameters fraction of limiting power density (FLPD) and maximum fraction of limiting power density (MFLPD). The licensee proposes to delete the definitions of FLPD and MFLPD from the Definitions section of the TS.

FLPD and MFLPD are GE terms. SPC uses Fuel Design Limiting Ratio to monitor LHGR. The proposed change makes the TS vendor independent by deleting the definitions for FLPD and MFLPD and placing them in the COLR. The definition of LHGR will include a vendor independent statement that LHGR is monitored by the ratio of LHGR to its fuel specific limit, as specified in the COLR. There is no change to the limiting condition for operation (LCO) for LHGR which currently requires that LHGR be maintained less than or equal to the LHGR limit specified in the COLR. These changes are acceptable. The proposed addition to the definition of LHGR was inadvertently not included in the proposed Unit 1 TS pages submitted by the licensee. This change was included in the proposed Unit 2 TS pages. The licensee has stated that the change is applicable to both units.

## 2.2 Critical Power Ratio (CPR)

LaSalle currently uses the GE critical power correlation called GEXL to calculate the CPR for the GE fuel bundles. The licensee proposes to revise the definition of CPR in section 1.9 of the TS by deleting reference to the GEXL correlation which is GE specific and replacing it with the term "approved CPR correlation". GEXL will no longer be used once SPC fuel is loaded. Instead, the SPC critical power correlation (ANFB) will be the correlation of record. The change to a non vendor-specific term does not change any TS requirements and is acceptable.

TS 3.2.3, Minimum Critical Power Ratio, requires that the MCPR be equal to or greater than the MCPR limit specified in the COLR. The MCPR operating limit specified in the COLR may be scram time dependent. TS 4.2.3 specifies scram times to be used to determine the applicable MCPR limit. The current TS requires that an average scram time of 0.86 seconds be used prior to performance of the initial scram time measurements or an average scram time may be used within 72 hours of the conclusion of each scram time surveillance. The current TS is based on GE methodology which uses a 20% scram insertion point as the reference for scram time surveillance. The TS maximum average scram time from the 20% scram insertion point is 0.86 seconds. The licensee proposes to revise the required scram times used to determine MCPR operating limits to be consistent with SPC methodology. SPC methods use the 5%, 20%, 50%, and 90% scram insertion points to determine the MCPR operating limit. The licensee proposes to revise TS 4.2.3 to delete the requirement to use 0.86 seconds prior to performance of the initial scram time measurements and instead use Technical Specification Scram Speeds (TSSS). TS 3.1.3.3 specifies the required average scram insertion times for the 5%, 20%, 50%, and 90% scram insertion points. The use of four scram times provides additional conservatism and is acceptable. In addition, rather than using the average scram insertion times from the surveillances, the proposed TS use the nominal scram speed (NSS) insertion times specified in the COLR if the surveillance results meet the NSS insertion times. SPC methods for analyzing the scram time dependence of the MCPR operating limit utilize cycle-specific nominal values. The process for determining the MCPR operating limits has been replaced with the SPC methods, including use of the NSS which is maintained in the COLR. The proposed change appropriately reflects the NRC-approved SPC



methodology and does not change the current requirement that MCPR meet the limits specified in the COLR. Therefore, the proposed change is acceptable.

Current TS 6.6.A.6.a.2 requires that core operating limits be established and documented in the COLR for MCPR before each reload cycle. The licensee proposes to delete the requirement that MCPR limits be based on a 20% scram insertion time and instead require that scram time dependent MCPR limits be established. As discussed above, the GE methodology which bases MCPR operating limits on the 20% scram insertion time is being replaced with SPC methodology which uses four scram insertion points. The change to TS 6.6.A.6.a.2 reflects this change and is acceptable.

The licensee also proposes to add to TS 6.6.A.6.a.2 the requirement to include the effects of analyzed equipment out of service in the COLR. The requirement is being added to TS 6.6.A.6.a.2 because SPC methodology performs the out of service analyses on a cycle-specific basis, the results of which are documented in the COLR before each reload. The proposed change accurately reflects the revised SPC methodology and is acceptable. The details of these analyses had previously been relocated from the bases section of the TS to the COLR by letter dated May 23, 1995.

Current TS 3.4.1.1 "Recirculation Loops" Action a.1.b requires that, with one recirculation loop in operation, within four hours, the MCPR safety limit must be increased by 0.01 to 1.08 per TS 2.1.2. The licensee proposes to delete the specific requirement to increase MCPR to 1.08. TS 2.1.2 requires that MCPR shall not be less than 1.08 with single recirculation loop operation. Therefore, the inclusion of the MCPR limit in TS 3.4.1.1 is repetitious and unnecessary and its deletion is acceptable.

### 2.3 Average Planar Linear Heat Generation Rate (APLHGR)

TS 3.2.1, Average Planar Linear Heat Generation Rate, requires that the APLHGR not exceed the limits specified in the COLR. This specification assures that the peak cladding temperature following the postulated design basis LOCA will not exceed the limit specified in 10 CFR 50.46. Current TS 3.4.1.1, Recirculation Loops, action a, requires certain actions to be taken when only one reactor coolant system recirculation loop is in operation. The licensee proposes to add action a.1.e to require that the APLHGR limit be reduced by the applicable single loop operation factor specified in the COLR. This requirement is specific to SPC fuel. The GE methodology does not require a specific single loop operation (SLO) factor to be applied because GE methodology, as documented in the COLR, uses multipliers based on core flow (SLO results in a reduction of 30 - 40% in total core flow) which have the same effect on APLHGR limits as the use of a SLO factor. The proposed requirement conservatively ensures that the peak cladding temperature for SLO is bound by the peak cladding temperature for two-loop operation and is acceptable.

## 2.4 Traversing In-core Probe (TIP)

TS 3.3.7.7 requires that the traversing in-core probe (TIP) system be operable for the purpose of calibrating LPRM detectors. The TIP system allows calibration of LPRM signals by correlating TIP signals to LPRM signals as the TIP is positioned in various radial and axial locations in the core. The current TS allows the use of substituted TIP data from symmetric channels if the control rod pattern is symmetric. SPC methods use a statistical check of TIP symmetry, which is an assumed parameter in their analysis methods. Therefore, this method needs to be incorporated into the LaSalle Core Operating Limits Report (COLR). The licensee proposes to relocate the requirements of TS 3/4.3.7.7 from the TS to the COLR.

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in a Limiting Condition for Operation (LCO), as follows: (1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of a primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic

safety assessment has shown to be significant to public health and safety. As a result, existing TS LCO requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents. The Commission amended 10 CFR 50.36 to codify and incorporate these four criteria (60 FR 36953).

The four criteria are discussed below:

**Criteria 1:** Installed instrumentation used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The TIP is used as a calibration tool for the LPRMs. It is not used for detecting and indicating degradation of the primary pressure boundary. Any leakage of the TIP tubing in the reactor pressure boundary would be indicated in the control room similar to any other primary leak (e.g., through increased drywell pressure or increased sump flow rates).

**Criteria 2:** A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The TIP system is used only as a calibration tool for the LPRMs and is not a condition of a Design Basis Accident or Transient analysis. Its function as a calibration tool for the LPRMs results in uncertainties which are included in the core monitoring methods.

**Criteria 3:** A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The TIP system's direct accident/transient function is the containment isolation function of the TIPs when they are penetrating primary containment. However, this system function is not related to the calibration function covered by the subject TS. The calibration function of the TIP which is required by the current TS is not a portion of the primary success path of safety sequence analysis.

**Criteria 4:** A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The licensee did not identify any probabilistic risk assessment concerns with the TIP system. The TIP system calibrates the LPRMs which are not safety related and are not required to actuate safety systems.

The TIP operability requirements and action statements will be relocated to the COLR which is required to be submitted to the NRC per TS 6.6.A.6. Any changes in the design or procedures for the TIP system will be evaluated for unreviewed safety questions per 10 CFR 50.59. Because this change meets the four criteria for relocating LCOs from the TS and changes to the relocated requirements will be adequately controlled, the relocation of TS 3/4.3.7.7 to the COLR is acceptable.

## 2.5 Reactivity Anomaly

TS 3.1.2 requires that the reactivity equivalence of the difference between the actual rod density and the predicted rod density be less than or equal to 1 percent delta k/k. This limit ensures that plant operation is maintained within the assumptions of the safety analyses. The licensee proposes to replace the words "rod density" with "critical control rod configuration" in the LCO and surveillance requirement. The current methodology utilizes a correlation between a change in rod density and core reactivity. The licensee is modifying its methodology to use a core monitoring code to monitor predicted Keff vs. actual Keff. This monitoring system uses critical control rod configuration as an input. The new methodology is a more direct measurement method and provides a more accurate estimate of the anomaly than the current method. Because the proposed change to the TS only revises the method of measuring the difference between predicted and monitored core reactivity and does not change the required limit, the change is acceptable.

## 2.6 Fuel Bundle Description

TS 5.3.1, Fuel Assemblies, provides a description of the fuel assemblies. The licensee proposes to expand this description consistent with Improved Standard Technical Specifications, NUREG-1434, and to better reflect the ATRIUM-9B design. The revised description includes discussion of the use of water rods or water boxes which is consistent with the SPC fuel design. The proposed TS includes the statement that a limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions, consistent with NUREG-1434. The proposed change accurately describes the SPC fuel design, is consistent with NUREG-1434, and does not affect any current TS requirements. Therefore, the proposed change is acceptable.

The proposed TS also states that limited substitutions of Zircalloy or ZIRLO or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. This change is consistent with NUREG-1434, and incorporates the guidance of Generic Letter 90-02, Supplement 1, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications." The GL provided flexibility in the repair of fuel assemblies containing damaged and leaking fuel rods by reconstituting the assemblies. The number and location of filler rod substitutions are limited to configurations for which applicable NRC approved codes and methods are valid and that have been shown by test or analyses to comply with all fuel safety design bases. To satisfy generic fuel design criteria as described in the Standard Review Plan, the filler rods

require thermal-hydraulic, neutronic, and mechanical analyses to demonstrate that inclusion of the filler rods in fuel assemblies with the specific configurations and core locations chosen for a specific fuel cycle is acceptable with respect to overall fuel performance and safety considerations. The proposed TS change complies with the guidance in GL 90-02, Supplement 1, will permit safe core configurations, and is acceptable.

## 2.7 Miscellaneous Changes

Changes are made to the index pages to reflect deleted definitions and deletion of the TIP TS as discussed above.

Numerous changes are made to the TS Bases to reflect the changes discussed above. In addition, a typographical error is corrected on page B 2-9 to agree with the UFSAR. The current bases page states that the IRM system terminates a low power control rod withdrawal error transient at 1% rated thermal power. The correct power level is 21%. The basis for the correction is the analysis in section 15.4.1.2 of the Updated Final Safety Analysis Report. Therefore, this change is acceptable.

The licensee proposes to revise TS 6.6.A.6.b to include references to the following topical reports which are used to determine the core operating limits.

- (1) ANFB Critical Power Correlation, ANF-1125(P)(A) and Supplements 1 and 2, Advanced Nuclear Fuels Corporation, April 1990.
- (2) Letter, Ashok C. Thadani (NRC) to R.A. Copeland (SPC), "Acceptance for Referencing of ULTRAFLOW™ Spacer on 9x9-IX/X BWR Fuel Design," July 28, 1993.
- (3) Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, XN-NF-524(P)(A) Revision 2, and Supplement 1 Revision 2, Supplement 2, Advanced Nuclear Fuels Corporation, November 1990.
- (4) COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis, ANF-913(P)(A), Volume 1, Revision 1 and Volume 1 Supplements 2, 3 and 4, Advanced Nuclear Fuels Corporation, August 1990.
- (5) HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option, ANF-CC-33(P)(A), Supplement 1 Revision 1; and Supplement 2, Advanced Nuclear Fuels Corporation, August 1986 and January 1991, respectively.
- (6) Advanced Nuclear Fuel Methodology for Boiling Water Reactors, XN-NF-80-19(P)(A), Volume 1, Supplement 3, Supplement 3 Appendix F,

and Supplement 4, Advanced Nuclear Fuels Corporation, November 1990.

- (7) Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads, XN-NF-80-19(P)(A), Volume 4, Revision 1, Exxon Nuclear Company, June 1986.
- (8) Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description, XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Company, January 1987.
- (9) Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, XN-NF-85-67(P)(A) Revision 1, Exxon Nuclear Company, September 1986.
- (10) Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel, ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, October 1991.
- (11) Volume 1 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Volume 2 - STAIF - A Computer Program for BWR Stability Analysis in the Frequency Domain, Code Qualification Report, EMF-CC-074(P)(A), Siemens Power Corporation, July 1994.
- (12) RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, XN-NF-81-58(P)(A), Revision 2 Supplements 1 and 2, Exxon Nuclear Company, March 1984.
- (13) XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2; Volume 1 Supplement 4, Advanced Nuclear Fuels Corporation, February 1987 and June 1988, respectively.
- (14) Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation, January 1993.
- (15) Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis, XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, Exxon Nuclear Company, Richland, WA 99352, March 1983.

- (16) Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, XN-NF-79-71(P)(A), Revision 2 Supplements 1, 2, and 3, Exxon Nuclear Company, March 1986.
- (17) Generic Mechanical Design Criteria for BWR Fuel Designs, ANF-89-98(P)(A), Revision 1 and Revision 1 Supplement 1, Advanced Nuclear Fuels Corporation, May 1995.
- (18) Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods," Revision 0, Supplements 1 and 2, December 1991, March 1992, and May 1992, respectively; SER letter dated March 22, 1993.

These additional topical reports are those used in SPC methodology and have been approved by the NRC for use at LaSalle. The staff finds this change acceptable because the use of identified NRC-approved methodologies will ensure that the values for cycle-specific parameters are determined consistent with applicable design bases and safety limits, and assist safe operation of the facility.

### 3.0 STATE CONSULTATION

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

### 4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 (61 FR 25699). The amendment also revises reporting requirements or record keeping requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and c(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

### 5.0 CONCLUSION

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

and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Skay

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