

ATTACHMENT 2

LIMERICK GENERATING STATION

Docket No. 50-353

License No. NPF-85

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

List of Attached Pages

Unit 2

xviii  
2-1  
B 2-1  
3/4 2-1  
3/4 2-8  
B 3/4 2-1  
B 3/4 2-2  
B 3/4 2-3  
B 3/4 2-5

INDEX

BASES

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<u>SECTION</u>	<u>PAGE</u>
<u>3/4.0 APPLICABILITY</u> .....	B 3/4 0-1
<u>3/4.1 REACTIVITY CONTROL SYSTEMS</u>	
3/4.1.1 SHUTDOWN MARGIN.....	B 3/4 1-1
3/4.1.2 REACTIVITY ANOMALIES.....	B 3/4 1-1
3/4.1.3 CONTROL RODS.....	B 3/4 1-2
3/4.1.4 CONTROL ROD PROGRAM CONTROLS.....	B 3/4 1-3
3/4.1.5 STANDBY LIQUID CONTROL SYSTEM.....	B 3/4 1-4
<u>3/4.2 POWER DISTRIBUTION LIMITS</u>	
3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE.....	B 3/4 2-1
LEFT INTENTIONALLY BLANK.....	B 3/4 2-3
3/4.2.2 APRM SETPOINTS.....	B 3/4 2-2
3/4.2.3 MINIMUM CRITICAL POWER RATIO.....	B 3/4 2-4
3/4.2.4 LINEAR HEAT GENERATION RATE.....	B 3/4 2-5
<u>3/4.3 INSTRUMENTATION</u>	
3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION.....	B 3/4 3-1
3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION.....	B 3/4 3-2
3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION.....	B 3/4 3-2
3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION.....	B 3/4 3-3
3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION.....	B 3/4 3-4
3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION.....	B 3/4 3-4
3/4.3.7 MONITORING INSTRUMENTATION	
Radiation Monitoring Instrumentation.....	B 3/4 3-4

## 2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS

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

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

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

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

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 for two recirculation loop operation and shall not be less than 1.08 for single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

#### ACTION:

With MCPR less than 1.07 for two recirculation loop operation or less than 1.08 for single recirculation loop operation and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

#### REACTOR COOLANT SYSTEM PRESSURE

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

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

#### ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

## 2.1 SAFETY LIMITS

### BASES

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## 2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principle 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

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. 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  $28 \times 10^3$  lb/hr, 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  $28 \times 10^3$  lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia 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.

## 3/4.2 POWER DISTRIBUTION LIMITS

### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

#### LIMITING CONDITION FOR OPERATION

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3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of axial location and AVERAGE PLANAR EXPOSURE shall be within limits based on applicable APLHGR limit values which have been determined by approved methodology for the respective fuel and lattice types for two recirculation loop operation. When hand calculations are required, the APLHGR for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) as shown in the CORE OPERATING LIMITS REPORT (COLR). During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the reduction factors shown in the COLR.

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

#### ACTION:

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

#### SURVEILLANCE REQUIREMENTS

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4.2.1 All APLHGRs shall be verified to be equal to or less than the limiting value:

- a. At least once per 24 hours,
- b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
- c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

## POWER DISTRIBUTION LIMITS

### 3/4.2.3 MINIMUM CRITICAL POWER RATIO

#### LIMITING CONDITION FOR OPERATION

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3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit times the  $K_f$ , both values shown in the CORE OPERATING LIMITS REPORT, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

$\tau_A$  = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65 \left( \frac{N_1}{\sum_{i=1}^n N_i} \right)^{1/2} (0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^n N_i \tau_i}{\sum_{i=1}^n N_i},$$

$n$  = number of surveillance tests performed to date in cycle,

$N_i$  = number of active control rods measured in the  $i^{th}$  surveillance test,

$\tau_i$  = average scram time to notch 39 of all rods measured in the  $i^{th}$  surveillance test, and

$N_1$  = total number of active rods measured in Specification 4.1.3.2.a.

#### APPLICABILITY:

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

### 3/4.2 POWER DISTRIBUTION LIMITS

#### BASES

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#### 3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10CFR50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 2) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 2) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 2. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10CFR50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits comply with the ECCS limits specified in 10CFR50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant.

The MAPLHGR limit as shown in the CORE OPERATING LIMITS REPORT is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the ECCS MAPLHGR limit.

Only the most limiting MAPLHGR values are shown in the CORE OPERATING LIMITS REPORT for multiple lattice fuel. Compliance with the specific lattice MAPLHGR operating limits, which are available in Reference 3, is ensured by use of the process computer.

The MAPLHGR limits shall be reduced to the value shown in the CORE OPERATING LIMITS REPORT times the two recirculation loop operation limit when in single recirculation loop operation. The constant factor shown in the CORE OPERATING LIMITS REPORT is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluation.

## POWER DISTRIBUTION LIMITS

### BASES

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#### 3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased neutron flux-upscale scram trip setpoint and flow biased neutron flux-upscale control rod block functions of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the Safety Limit MCPR or that  $> 1\%$  plastic strain does not occur in the degraded situation. The scram and rod block setpoints are adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

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## POWER DISTRIBUTION LIMITS

### BASES

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#### MINIMUM CRITICAL POWER RATIO (Continued)

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at RATED THERMAL POWER and rated thermal flow.

The  $K_f$  factors shown in CORE OPERATING LIMITS REPORT are conservative for the General Electric Boiling Water Reactor plant operation because the operating limit MCPRs of Specification 3.2.3 are greater than the original 1.20 operating limit MCPR used for the generic derivation of  $K_f$ .

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 startup 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

This 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.

#### References:

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20563, November 1975.
2. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (latest approved revision).
3. "Basis of MAPLHGR Technical Specifications for Limerick Unit 2," NEDC-31930P (as amended).
4. Deleted.
5. Increased Core Flow and Partial Feedwater Heating Analysis for Limerick Generating Station Unit 2 Cycle 1, NEDC-31578P, March 1989 including Errata and Addenda Sheet No. 1 dated May 31, 1989.
6. General Electric Boiling Water Reactor Extended Load Line Limit Analysis for Limerick Generating Station Unit 2, Cycle 1, NEDC-31677P, March 1989.