ATTACHMENT 2

LIMERICK GENERATING STATION

UNIT 2

DOCKET NO. 50-353

LICENSE NO. NPF-85

TECHNICAL SPECIFICATIONS CHANGE REQUEST

NO. 96-18-2

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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~~ 1.10 for two recirculation loop operation and shall not be less than ~~1.08~~ 1.12 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~~ 1.10 for two recirculation loop operation or less than ~~1.08~~ 1.12 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.

BASES

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~~ 1.10 for two recirculation loop operation and ~~1.08~~ 1.12 for single recirculation loop operation. MCPR greater than ~~1.07~~ 1.10 for two recirculation loop operation and ~~1.08~~ 1.12 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 x 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 x 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.

ATTACHMENT 3

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NO. 96-18-2

Letter, R. M. Butrovich (GE) to H. J. Diamond (PECO Energy),

"Limerick Unit 2 Safety Limit
MCPR Revision," dated July 7, 1996

GE Nuclear Energy

Richard M. Butrovich

July 7, 1996
RMB:96-145

Mr. H. J. Diamond, Director
Fuel & Services Division
PECO NUCLEAR
965 Chesterbrook Boulevard
Wayne, PA 19087-5691

SUBJECT: **Limerick Unit 2 Safety Limit MCPR Revision**

REFERENCE:
1. Letter, R. M. Butrovich to H. J. Diamond, "Generic GE11 Safety Limit MCPR Calculation", April 2, 1996
2. Letter, R. M. Butrovich to H. J. Diamond, "Limerick Unit 2 Safety Limit MCPR", May, 21, 1996.
3. Letter, R. M. Butrovich to H. J. Diamond, "Limerick 2 Reload 3 (Cycle 4) SRLR Revision 1", June 14, 1996.

Dear Hugh:

Reference 1 advised PECO Nuclear of discoveries related to the methodology used by GE to calculate SLMCPR that indicated the generic SLMCPR may not always yield the most conservative result. GE performed a plant unique evaluation for Limerick Unit 2 and reported the results to PECO Nuclear in reference 2. The SLMCPR for Limerick Unit 2 Cycle 4 went from 1.07 (GE11 Generic) to 1.10 (Cycle specific). The single loop operation SLMCPR is 1.11. Reference 3 provided a revised Supplemental Reload Licensing Report for Limerick Unit 2 Cycle 4. PECO Nuclear should submit a technical specification revision indicating the revised cycle specific SLMCPR.

If you have any questions, please give me a call.

Very truly yours,

R. M. Butrovich