10 CFR 50.90

PHILADELPHIA ELECTRIC COMPANY

NUCLEAR GROUP HEADQUARTERS

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December 9, 1993

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

STATION SUPPORT DEPARTMENT

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Limerick Generating Station, Units 1 and 2
Operating License Change Request 93-24-0

Gentlemen:

Philadelphia Electric Company (PECo) hereby requests a change to Operating License Nos. NPF-39 and NPF-85 and their corresponding Appendices A, for Limerick Generating Station (LGS), Units 1 and 2 respectively. The proposed changes to the Operating Licenses and their corresponding Appendices A reflect the planned implementation of the Power Rerate Program at LGS Units 1 and 2, and the corresponding increase in the authorized maximum reactor core power level by five percent to 3458 megawatts thermal (MWt) from the current limit of 3293 MWt.

Attachment 1 to this letter describes the proposed changes, and contains information supporting a finding that the proposed changes do not involve a Significant Hazards Consideration and information supporting an Environmental Assessment. Attachment 2 contains the Operating License and Appendix A pages showing the proposed changes. Attachment 3, NEDC-32225P, "Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 & 2," dated September 1993, contains the safety analysis prepared by General Electric (GE) to support this Change Request and the implementation of Power Rerate at LGS, Units 1 and 2. The analyses and evaluations supporting these proposed changes were completed using the guidelines in GE Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," dated May 1992. The resolution of generic issues associated with power uprate was addressed in GE Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991. These reports are proprietary and have been submitted separately to the NRC by GE. The NRC reviewed and approved these Topical Reports by letters from the NRC to GE dated September 30, 1991 and July 31, 1992. The safety analysis in Attachment 3 complies with the guidelines of NEDC-31897P-A and supplements the generic evaluations of NEDC-31984P with LGS specific information.

Attachment 3 contains information proprietary to GE. GE requests that the document be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4). An affidavit supporting this request in accordance with 10 CFR 2.790(b)(1) is provided with Attachment 3.

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We are requesting that, if approved, the amendments be issued on December 27, 1994, for LGS Unit 2, and December 26, 1995, for LGS Unit 1, and be made effective within 30 days after the issue date.

If you have any questions or require additional information, please contact us.

Very truly yours,

RM Krich for

G. A. Hunger, Jr., Director
Licensing Section

Attachments

cc: T. T. Martin, Administrator, Region I, USNRC w/ attachments
N. S. Perry, USNRC Senior Resident Inspector, LGS "
W. P. Dornsife, Director, Bureau of Radiological Protection "

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

D. R. Helwig, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company; the Applicant herein; that he has read the foregoing Application for Amendment of Facility Operating License Nos. NPF-39 and NPF-85, Operating License Change Request No. 93-24-0, for the Power Rerate Program to be implemented at Limerick Generating Station, Units 1 and 2, and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

D R Helwig

Vice President

Subscribed and sworn to

before me this 9th day

of December 1993.

Erica A. Santon

Notary Public

Notarial Seal
Erica A. Santon, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires July 10, 1995

ATTACHMENT 1

LIMERICK GENERATING STATION
Units 1 and 2

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

OPERATING LICENSE CHANGE REQUEST

"Power Rerate Program for Limerick Generating Station, Units 1 and 2"

Supporting Information for Changes - 24 pages

Philadelphia Electric Company (PECo), licensee under Facility Operating License Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2, requests that these licenses be amended as proposed herein to reflect the Power Rerate Program to be implemented at LGS, Units 1 and 2, specifically to increase the maximum reactor core power level by five percent (5%), to 3458 megawatts thermal (MWt) from the current limit of 3293 MWt.

The analyses and evaluations supporting these proposed changes were completed using the guidelines in General Electric (GE) Topical Report NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," dated May 1992. This Topical Report was reviewed and approved by the NRC, by letter to GE, dated September 30, 1991. Resolution of generic issues associated with power uprate was addressed in GE Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991. This Topical Report was reviewed and approved by the NRC, by letter to GE, dated July 31, 1992.

An increase in electrical output is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Continuing improvements in the analytical techniques (i.e., computer codes and data) based on several decades of Boiling Water Reactor (BWR) safety technology, plant performance feedback, and improved fuel and core design, have resulted in a significant increase in the margin between calculated safety analysis results and the licensing limits. This available safety analysis margin, combined with the excess capability of as-designed equipment, systems, and components, provides the potential for an increase of 5% in the full power rating of a plant without the need to perform major Nuclear Steam Supply System (NSSS) or Balance-of-Plant (BOP) hardware modifications. The full power level can be increased safely, and the installed systems and equipment are capable of performing required functions at the rerated conditions. The method for achieving higher power is to extend the reactor core power-flow map by increasing reactor core flow along existing flow control lines. However, there will not be an increase in the maximum recirculation flow limit over the pre-rerate value. Most of the original safety analyses, such as the transient (i.e., abnormal operating events) analyses, were based on 105% steam flow, which coincides closely with the steam flow at the proposed rerated full power level.

The safety analysis prepared by GE to support this Change Request and the implementation of the Power Rerate Program at LGS, Units 1 and 2, is provided in Attachment 3, NEDC-32225P, "Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 & 2," dated September 1993, and demonstrates that the LGS, Units 1 and 2 can operate safely with the proposed 5% increase of the reactor thermal power and an associated 40 psi increase in the operating reactor vessel pressure, with a corresponding increase in main turbine inlet steam flow and the corresponding increases of the flow, temperature, pressure, and capacity required in supporting systems and components at these rerated conditions. This safety analysis has been performed taking into account the current 24 month refueling cycles for LGS, Units 1 and 2, and the implementation of Average Power Range Monitor-Rod Block Monitor Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) at LGS, Units 1 and 2, prior to implementation of power rerate. By letter dated August 27, 1993, PECo submitted Technical Specifications Change Request (TSCR) No. 92-08-0, proposing the ARTS/MELLLA TS changes for LGS, Units 1 and 2, to the NRC.

This Operating License Change Request for LGS, Units 1 and 2, provides a discussion and description of the proposed changes, a safety assessment, information supporting a finding of No Significant Hazards Consideration, and information supporting an Environmental Assessment.

We request that, if approved, the amendments be issued on December 27, 1994, for LGS Unit 2, and December 26, 1995, for LGS Unit 1, and be made effective within 30 days after the issue date.

I. Discussion and Description of the Proposed Changes

A. The proposed Operating License and Appendix A changes associated with the planned implementation of the Power Rerate Program at Limerick Generating Station (LGS), Units 1 and 2 are as follows.

Operating License (OL)

a) Revise Operating License "Maximum Power Level," Page 3.

OL Appendix A "Technical Specifications" (TS)

a) Definitions

i) Revise the Rated Power value, TS Section 1.0, Page 1-6.

b) Reactor Protection System Instrumentation Setpoints

i) Revise the Average Power Range Monitor (APRM) Flow Biased "Neutron Flux-Upscale" trip setpoint and allowable value during two recirculation loop operation, TS Table 2.2.1-1, Page 2-4.

ii) Revise the APRM Flow Biased "Neutron Flux- Upscale" trip setpoint and allowable value during single recirculation loop operation, TS Table 2.2.1-1, Page 2-4.

iii) Revise the trip setpoint and allowable value for "Reactor Vessel Steam Dome Pressure - High," TS Table 2.2.1-1, Page 2-4.

c) Standby Liquid Control System (SLCS)

i) Revise the Surveillance Requirement for the SLCS pump pressure, TS Section 4.1.5, Page 3/4 1-20.

d) Isolation Actuation Instrumentation Setpoints

i) Revise the "Main Steam Line Flow - High" trip setpoint and allowable value, TS Table 3.3.2-2, Page 3/4 3-18.

ii) Revise the "Reactor Water Cleanup Area Temp - High" trip setpoint and allowable value, TS Table 3.3.2-2, Page 3/4 3-19.

iii) Revise the "High Pressure Coolant Injection (HPCI) Steam Line Δ Pressure - High" trip setpoint and allowable value, TS Table 3.3.2-2, Page 3/4 3-19.

iv) Revise the "Reactor Core Isolation Cooling (RCIC) Steam Line Δ Pressure - High" trip setpoint and allowable value, TS Table 3.3.2-2, Page 3/4 3-20.

e) Anticipated Transients Without Scram (ATWS) Recirculation Pump Trip System Instrumentation Setpoints

i) Revise the "Reactor Vessel Pressure - High" trip setpoint and allowable value, TS Table 3.3.4.1-2, Page 3/4 3-44.

f) Control Rod Block Instrumentation Setpoints

i) Revise the APRM Flow Biased "Neutron Flux - Upscale" trip setpoint and allowable value during two recirculation loop operation, , TS Table 3.3.6-2, Page 3/4 3-60.

ii) Revise the APRM Flow Biased "Neutron Flux - Upscale" trip setpoint and allowable value during single recirculation loop operation, , TS Table 3.3.6-2, Page 3/4 3-60.

g) Recirculation System

i) Revise the Limiting Condition for Operation (LCO) Action a.1.b for thermal power limit for Single Loop Operation, TS Section 3.4.1.1, Page 3/4 4-1.

ii) Revise the Surveillance Requirement for thermal power limit for Single Loop Operation, TS Section 4.4.1.1.4, Page 3/4 4-2.

iii) Revise the Thermal Power versus Core Flow plot, TS Figure 3.4.1.1-1, Page 3/4 4-3.

h) Reactor Coolant System Safety Relief Valves

i) Revise the LCO for all Safety Relief Valve setting values, TS Section 3.4.2, Page 3/4 4-7.

i) Reactor Coolant System Pressure/Temperature Limits

i) Revise the "Minimum Reactor Pressure Vessel Metal Temperature versus Reactor Vessel Pressure" plot, TS Figure 3.4.6.1-1, Page 3/4 4-20.

ii) Revise the LCO to also address the use of curve A' from the revised TS Figure 3.4.6.1-1, TS Section 3.4.6.1, Page 3/4 4-18 (Unit 1 only).

iii) Revise the Surveillance Requirement to address the use of curve A' from the revised TS Figure 3.4.6.1-1, TS Section 4.4.6.1.1, Page 3/4 4-18 (Unit 1 only).

j) Reactor Coolant System - Reactor Vessel Steam Dome

i) Revise the LCO for reactor vessel steam dome pressure, TS Section 3.4.6.2, Page 3/4 4-22.

ii) Revise the Surveillance Requirement for reactor vessel steam dome pressure, TS Section 4.4.6.2, Page 3/4 4-22.

k) Emergency Core Cooling Systems (ECCS)

i) Revise the Surveillance Requirement for reactor vessel pressure and HPCI turbine inlet pressures, TS Section 4.5.1 Page 3/4 5-4.

l) Secondary Containment

i) Revise the Surveillance requirement for drawdown time, TS Section 4.6.5.1.1, Page 3/4 6-46.

m) Standby Gas Treatment System (SGTS)

i) Revise the Surveillance requirement for drawdown time, TS Section 4.6.5.3, Page 3/4 6-54.

n) Reactor Core Isolation Cooling (RCIC) System

i) Revise the Surveillance Requirement for RCIC pump turbine inlet pressure, TS Section 4.7.3, Page 3/4 7-9.

o) Reactor Coolant System Pressure/Temperature Limits - TS Bases

i) Revise the Pressure/Temperature Limits Bases section to account for the shift in Reference Temperature Nil-Ductility Transition (RT_{NDT}) and updated Effective Full Power Year (EFPY) values from curves A/A', B, and C for Unit 1, and to account for the assumed shift in RT_{NDT} and new EFPY for Unit 2. TS Figure 3.4.6.1-1, TS Bases 3/4.4.6, Page B 3/4 4-5.

ii) Revise the Reactor Vessel Toughness Bases Table, TS Bases Table B 3/4.4.6-1, Page B 3/4 4-7.

iii) Revise the "Fast Neutron Fluence as a function of Service Life," TS Bases Figure B 3/4.4.6-1, Page B 3/4 4-8.

p) ECCS - TS Bases

i) Revise the HPCI related reactor pressure, TS Bases 3/4.5.1 and 3/4.5.2, Page B 3/4 5-1.

q) Primary Containment Systems - TS Bases

i) Revise the Primary Containment Bases for the calculated peak accident pressure, TS Bases 3/4.6.1.2, Page B 3/4 6-1.

ii) Clarify text and delete the specific value of the maximum calculated pressure value for the Primary Containment Structural Integrity, TS Bases 3/4.6.1.5, Page B 3/4 6-2.

iii) Clarify text and delete the specific value of the calculated containment peak pressure for the Drywell and Suppression Chamber Internal Pressure, TS Bases 3/4.6.1.6, Page B 3/4 6-2.

iv) Revise the value of the Drywell and Suppression Chamber Internal Pressure, TS Bases 3/4.6.1.6, Page B 3/4 6-2.

r) Depressurization Systems - TS Bases

i) Revise the Bulk Water Temperature to an initial suppression chamber water temperature of 95°F, TS Bases 3/4.6.2, Page B 3/4 6-3.

s) Secondary Containment Systems - TS Bases

i) Revise the drawdown time, TS Bases 3/4.6.5, Page B 3/4 6-5 (for Unit 1 and Unit 2), and Page B 3/4 6-6 (for Unit 2 only).

t) Reactor Coolant System

i) Revise nominal reactor vessel steam dome saturation temperature, TS Section 5.4.2, Page 5-8.

B. The proposed Operating License (OL) and Appendix A changes specified above for LGS, Units 1 and 2, have been grouped by categories and are discussed below. This discussion is a summary of information provided in the safety analysis provided in Attachment 3, NEDC-32225P, "Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 & 2," dated September 1993.

1. Revise the Average Power Range Monitor (APRM) equation to reflect the rated thermal power increase of 5%.

OL, Appendix A:

TS Table 2.2.1-1, pg. 2-4, and
TS Table 3.3.6-2, pg. 3/4 3-60.

The APRM signals will be recalibrated to the rerated maximum power level, and the percentage setpoints will be lowered. Since the Maximum Extended Load Line Limit (MELLL) of the reactor core power-flow map region will be changed as a result of the proposed power rerate, the APRM flow-biased control rod block and reactor SCRAM setpoints will be changed accordingly. These changes will maintain the same margin from the upper limit of the MELLL region of the power-flow map to assure that the change is conservative.

The current Rod Block Monitor (RBM) setpoints were assumed in the Rod Withdrawal Error (RWE) transient analysis at the rerated maximum power level. This analysis demonstrates that RBM performance at the rerated power conditions is adequate to ensure that local power excursions within the reactor core due to any potential RWE are maintained within acceptable limits.

Power rerate will have little effect on the Intermediate Range Monitors (IRMs) overlap with the Source Range Monitors (SRMs) and the APRMs. Using normal plant procedures, the IRMs will be adjusted, as required, so that overlap with the SRMs and APRMs remains adequate. No change is needed in the APRM downscale setting.

The neutronic life of the Local Power Range Monitor (LPRM) detectors and radiation level of the Traversing Incore Probe (TIP) may be affected slightly due to the higher power level. Therefore, a slight increase in change out frequency (i.e., approximately 5%) will have to be made for the LPRM detectors. The TIP detectors' lifetime will not be severely reduced due to its short and intermittent use.

2. Revise the values of the Safety Relief Valves (SRVs) settings by increasing the value of the setpoint by the increase in nominal operating pressure (i.e., 40 psi) at the rerated maximum power level.

   OL, Appendix A:

   TS Section 3.4.2, pg. 3/4 4-7.

   The installed SRVs have been determined to be acceptable under the rerated operating conditions. The SRVs setpoints will, however, be reset to higher pressure values. PECo has determined that the SRVs have the capability to accommodate operation at the rerated maximum power level.

3. Revise Reactor Core Isolation Cooling (RCIC) system operating requirements to reflect rerate conditions.

   OL, Appendix A:

   TS Table 3.3.2-2, pg. 3/4 3-20, and
   TS Section 4.7.3, pg. 3/4 7-9.

   The RCIC system provides core cooling when the Reactor Pressure Vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than maximum allowable for initiation of a low pressure core cooling system. The generic evaluation documented in Section 4.2.2 of the GE Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprates," dated July 1991, examined RCIC system pump and drive turbine operational requirements for plant operation with an SRV setpoint increase of about 40 psi above nominal RPV pressure, and concludes that the RCIC system is capable of delivering its design flow for RPV pressure increases up to 40 psi above nominal operating pressure. As the LGS, Units 1 and 2 RPV operating and SRV setpoint pressures will increase by 40 psi, the evaluation of the RCIC system adequacy under rerated power conditions is consistent with the bases and conclusions in NEDC-31984P.

The RCIC system pump turbine has the capacity to develop the horsepower and speed that will be required by the pump to meet the new RPV pressure conditions. The increase in pump and turbine speed requires a new rated speed of 4,575 rpm. This speed is below the maximum continuous operating speed specified by the pump and turbine manufacturers. The increased turbine rated speed will require that the overspeed trip margin be reduced in order to maintain the maximum trip speed below the limit specified by the manufacturer.

4. Revise High Pressure Coolant Injection (HPCI) system operating requirements to reflect rerate conditions.

   OL, Appendix A:

   TS Table 3.3.2-2, pg. 3/4 3-19,
   TS Section 4.5.1, pg. 3/4 5-4, and
   TS Bases 3/4.5.1 and 3/4.5.2, pg. B 3/4 5-1.

   The HPCI System was determined to have the capability to deliver its design steady state flow of 5,600 gpm at rerated power conditions. These conditions are based on a proposed SRV setpoint increase of 40 psi, accounting for the current SRV allowable setpoint tolerance of ±1%.

   The HPCI system pump turbine is capable of operating at the increased steam supply pressures and temperatures resulting from power rerate operating conditions. The pump is capable of delivering sufficient head to meet the new injection pressure requirements while maintaining an adequate system operating margin.

   The remainder of the system components were determined not to be impacted significantly by power rerate conditions because of the small increase in operating pressure and/or temperature.

5. Revise the Technical Specifications (TS) to reflect the increase in nominal operating pressure (i.e., 40 psi) at the rerated maximum power level.

   OL, Appendix A:

   TS Table 2.2.1-1, pg. 2-4, and
   TS Section 3.4.6.2, pg. 3/4 4-22.

   As a result of the RPV pressure increase associated with power rerate, the analytical limit for the reactor SCRAM at high RPV pressure will be increased accordingly. During a pressure increase transient not terminated by a direct SCRAM (e.g., main turbine stop valve closure) signal or high reactor core flux SCRAM, the high RPV pressure SCRAM will terminate the transient. The high RPV pressure SCRAM signal setting values will be maintained slightly above the

RPV maximum normal operating pressure and below the specified analytical limit. This setting permits normal operation without causing an inadvertent SCRAM, yet provides adequate margin to the maximum allowable RPV pressure.

The design pressure of the reactor vessel and reactor pressure coolant boundary remains at 1,250 psig. The American Society for Mechanical Engineers (ASME) Code allowable peak pressure is 1,375 psig (i.e., 110% of design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a Main Steam Isolation Valve (MSIV) closure with a failure of the MSIV position SCRAM. At rerated operating conditions, the peak RPV bottom head pressure remains below the 1,375 psig ASME limit.

6. Revise Standby Liquid Control System (SLCS) operating requirements to reflect power rerate conditions.

   OL, Appendix A:

   TS Section 4.1.5, pg. 3/4 1-20.

   The ability of the SLCS boron solution to achieve and maintain safe shutdown is not a direct function of core thermal power. However, due to increased fuel loading for higher power and extended operating cycle, the required concentration of boron may change. The shutdown capability (i.e., boron concentration) of the SLCS is re-evaluated for each core reload to ensure sufficient shutdown margin is available. In addition, the capability of the SLCS to provide its backup shutdown function has been shown to not be affected by power rerate since the small increase in RPV pressure due to the increase of the Safety Relief Valve (SRV) setpoints does not impact the SLCS positive displacement pumps' ability to attain rated injection flow to the RPV. Therefore, the capability of the SLCS to provide its backup reactor shutdown function is not affected by power rerate conditions.

7. Revise the secondary containment drawdown time to reflect the Standby Gas Treatment System (SGTS) safety analyses for power rerate conditions.

   OL, Appendix A:

   TS Sections 4.6.5.1.1 and 4.6.5.3, pp. 3/4 6-46 and 3/4 6-54, and TS Bases 3/4.6.5, pp. B 3/4 6-5 and B 3/4 6-6 (Unit 2 only).

   The secondary containment was reviewed for the impact of the rerate power conditions on the ability of the SGTS to drawdown and maintain the secondary containment at a negative pressure. The SGTS has sufficient capacity for rerate power conditions, but the Technical Specifications (TS) Surveillance Requirement drawdown test acceptance time will be increased by 5 seconds to accommodate higher heat loads resulting from rerated operating conditions.

8. Revise Reactor Coolant System Pressure/Temperature limits to reflect the reactor pressure vessel fracture toughness analyses at power rerate conditions.

   OL, Appendix A:

   TS Sections 3.4.6.1 and 4.4.6.1.1, pg. 3/4 4-18 (Unit 1 only),
   TS Figure 3.4.6.1-1, pg. 3/4 4-20,
   TS Bases B 3/4.4.6, pg. B 3/4 4-5,
   TS Bases Table B 3/4.4.6-1, pg. B 3/4 4-7, and
   TS Bases Figure B 3/4.4.6-1, pg. B 3/4 4-8.

   A conservative evaluation shows that for power rerate conditions there is a proportional increase in the reactor vessel inner radius peak neutron fluence. The increased fluence causes an increase in the Adjusted Reference Temperature (ART) after 32 Effective Full Power Years (EFPYs). This translates to an ART increase of approximately 3°F and 2°F for Unit 1 and Unit 2, respectively. As a result of this change, new Pressure Temperature (PT) curves are required.

   As well as affecting the ART, increased neutron fluence affects the Upper Shelf Energy (USE) prediction at end of the RPV design life. Based on the electroslag weld and plate surveillance USE values, the RPV beltline material USE will remain above 50 ft-lbs.

   In addition, since the higher neutron fluence is used to evaluate the reactor vessel against the requirements of 10CFR50, Appendix G, the results show the reactor vessel remains in compliance with this regulation and thus, operation at power rerate conditions will not have an adverse effect on the reactor vessel fracture toughness.

9. Revise various instrumentation trip setpoints and allowable values (i.e., Main Steam Line Isolation Flow - High, Reactor Water Cleanup System Area Temperature - High, and Anticipated Transients Without Scram (ATWS) Recirculation Pump Trip, Reactor Vessel Pressure - High) to reflect rerate conditions.

   OL, Appendix A:

   TS Table 3.3.2-2, pp. 3/4 3-18 and 3/4 3-19, and
   TS Table 3.3.4.1-2, pg. 3/4 3-44.

   The Main Steam Line Isolation Flow instrumentation will be recalibrated for the higher steam flow condition. This ensures sufficient margin to the trip setpoint exists to allow for normal plant testing of the MSIVs and turbine stop valves. The analytical limit for rerate conditions remains at 140% of the rerated steam flow.

   Ambient temperatures in the Reactor Water Cleanup (RWCU) System regenerative heat exchanger room may increase by 4°F. These temperature increases do not adversely impact component operation, although in some areas, the leak detection isolation temperature setpoint will be raised to preclude spurious isolations.

The ATWS high reactor coolant system pressure Recirculation Pump Trip (RPT) setpoint increases for power rerate conditions. The major consideration for this pressure setpoint is the increase in peak reactor coolant system pressure during a hypothetical ATWS event because of the higher initial power value. The effects of increasing safety relief valve and ATWS-RPT setpoints have been evaluated. These evaluations found a resulting increase in the peak reactor vessel bottom head pressure. However, the calculated peak pressure will remain below 1,375 psig for ATWS events.

10. Editorial changes

    a. Revise the value of the current rated thermal power (i.e., 3293 MWt) to the rerated power level (i.e., 3458 MWt).

       OL, pg. 3, and
       OL, Appendix A:

       TS Section 1.0, pg. 1-6.

    b. Revise TS Figure 3.4.1.1-1 to reflect power rerate conditions.

       OL, Appendix A:

       TS Figure 3.4.1.1-1, pg. 3/4 4-3.

    c. Revise the maximum allowable thermal power during single recirculation loop operation to reflect power rerate conditions, accounting for Maximum Extended Load Line Limit (MELLL) operation.

       OL, Appendix A:

       TS Sections 3.4.1.1 and 4.4.1.1.4, pp. 3/4 4-1 and 3/4 4-2.

    d. Revise and clarify text for the primary containment TS Bases to reflect the containment analysis for power rerate conditions.

       OL, Appendix A:

       TS Bases 3/4.6.1.2, pg. B 3/4 6-1, and
       TS Bases 3/4.6.1.5 and 3/4.6.1.6, pg. B 3/4 6-2.

    e. Revise the suppression pool bulk water temperature after initial reactor blowdown, to reflect the results of the containment analysis for power rerate conditions.

       OL, Appendix A:

       TS Bases 3/4.6.2, pg. B 3/4 6-3.

    f. Revise the nominal reactor steam dome saturation temperature to reflect rerate conditions.

OL, Appendix A:

TS Section 5.4.2, pg. 5-8.

Additional discussions of various systems, structures, and components that have been evaluated for rerated conditions but do not involve TS changes, are provided in Attachment 3, NEDC-32225P.

II. Safety Assessment of the Proposed Changes

A. Increasing the power level of Limerick Generating Station (LGS), Units 1 and 2, by 5% to 3458 MWt can be done safely within certain limits. Several light water reactors have already been rerated worldwide, including 17 Boiling Water Reactors (BWRs) in the United States, Switzerland, and Spain.

Most of the original safety analyses, such as the transient (i.e., abnormal operating events) analyses, were based on 105% of full steam flow, which coincides with about the same power level as is requested for the power rerate of the maximum power for LGS, Units 1 and 2. Those safety analyses exceeded the requirement to perform analyses at 102% of full power by about 2.2%. The power dependent safety analyses summarized below are based on 102% of rerated full power. Therefore, this safety assessment justifies only about an additional 3% power increase over the original analyzed power level.

An increase in electrical output of a BWR plant is accomplished primarily by generation and supply of higher steam flow to the turbine generator. LGS, like most BWR plants, as originally licensed, has an as-designed equipment and system capability to accommodate a steam flow rate at least 5% above the original rating. In addition, continuing improvements in the analytical techniques (i.e., computer codes and data) based on several decades of BWR safety technology, plant performance feedback, and improved fuel and core designs, have resulted in a significant increase in the difference between calculated safety analysis results and the licensing limits. This available safety analysis margin, combined with the excess capability of as-designed equipment, systems, and components, provide BWR plants with the potential for an increase in their full power rating of between 5% and 10% without major hardware modifications and with no significant increase in the postulated hazards presented by the plant as approved by the NRC at the original license stage.

NEDC-32225P, "Power Rerate Safety Analysis Report for Limerick Generating Station Units 1 & 2," dated September 1993, is provided in Attachment 3 and contains the safety analysis prepared by General Electric (GE) to support the proposed Operating License (OL) changes and implementation of power rerate at LGS, Units 1 and 2. This analysis is based on 24 month refueling cycles. The analyses and evaluations supporting power rerate changes were completed using the guidelines in GE Topical Report, NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," dated May 1992. Resolution of generic issues associated with power uprate was addressed in GE Topical Report, NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991, and LGS specific evaluations are provided in NEDC-32225P. The NRC reviewed and approved these Topical Reports by letters from the NRC to GE dated September 30, 1991 and July 31, 1992.

Plant performance and responses to hypothetical accidents and transients have been evaluated for LGS, Units 1 and 2 for a power rerate of 5% of full power. This safety assessment summarizes the information provided in NEDC-32225P (i.e., the safety significant plant reactions to events analyzed during the licensing the Units and potential effects on various margins of safety).

LGS, Units 1 and 2 were originally licensed for operation at 100% thermal power (i.e., 3293 MWt). The original safety analysis basis was that the reactor would be operating continuously at a power level at least 1.02 times the licensed power level; however, many of the original analyses had already been performed for the 105% full steam flow conditions. The rerate power level included in this evaluation is a 5% thermal power rerate. Therefore, a 5% power rerate is analyzed at a power level at least 1.02 x 1.05 x (original licensed power level), or 1.02 x (rerated power level).

Rerate	FRACTION OF RATED POWER	POWER	ANALYSIS***
0	1.0	X*	≥1.02X
5%	1.05	1.05X=Y	≥1.071X
5%	1.05	Y**	≥1.02Y

* Original power level (3293 MWt)
** Rerated power level (3458 MWt)
*** Some analyses are performed at 100% rated power, because the 2% power factor is already accounted for in the analysis methods.

The above rerate analysis basis assures that the power dependent margins prescribed by the applicable regulations will be maintained by meeting the appropriate regulatory criteria. NRC accepted computer codes and calculational techniques were used to perform the calculations that demonstrate that the stipulated criteria are met. Similarly, margin specified by application of the pertinent sections of the American Society of Mechanical Engineers (ASME) design rules will be maintained, as will other margins assuring that the proper criteria are used to judge the acceptability of the plant. Environmental margins will be maintained by not increasing any of the present limits for releases such as residual chloride concentrations in the station outfall or plant vent radiological limits, as a consequence of power rerate.

No change is required in the basic fuel design to achieve the rerated power levels or to meet the plant licensing limits. The current fuel operating limits will continue to be met at the rerated power level. Analyses for each fuel reload will continue to meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II," or otherwise approved in the Tech al Specifications (TS).

The plant's design concept includes a variety of ways to , water into the reactor vessel to mitigate postulated events. These cooling water sources will maintain reactor core integrity by providing adequate water for reactor core cooling. Consequently, there are high and low pressure, high and low volume, safety- and nonsafety-related means of delivering water to the reactor vessel. These means include the feedwater and/or condensate pumps, the Low Pressure Coolant Injection (LPCI) system and Core Spray (CS) system pumps, the High Pressure Coolant Injection system (HPCI) pump, the Reactor Core Isolation Cooling system (RCIC) pump, the Standby Liquid Control System (SLCS) pumps, and the Control Rod Drive (CRD) pumps. Many of these diverse water supply means are redundant in equipment and also redundant in system (e.g., there are several LPCI and CS pumps and completely redundant piping systems).

Power rerate does not result in an increase or decrease in the available water sources, nor does it change the selection of those assumed to function in the safety analyses. NRC approved methods were used for analyzing the performance of the Emergency Core Cooling Systems (ECCS) during Loss of Coolant Accidents (LOCAs). Power rerate results in a small (i.e., 5%) increase in reactor decay heat, and thus, the core cooling time to reach cold shutdown conditions is increased. However, the existing cooling capacity can bring the plant to cold shutdown within 24 hours.

Design basis accidents (DBAs) are very low probability hypothetical events whose characteristics and consequences are used in the design of the plant, so that the plant can mitigate their consequences to within acceptable regulatory limits. For BWR licensing evaluations, capability is demonstrated for coping with the range of hypothetical pipe break sizes in the largest recirculation, steam, and feedwater lines, a postulated break in one of the ECCS lines, and even down to breaks the size of instrument lines. This break range bounds the full spectrum of large and small high energy line breaks, and the success of the plant systems in mitigating these events while accommodating a single active equipment failure in addition to the postulated LOCA. Several of the most significant licensing assessments are made using these LOCA conditions. These assessments are as follows.

- Challenges to fuel or the ECCS performance analyses in accordance with the requirements and acceptance criteria of 10CFR50.46 and 10CFR50, Appendix K, wherein the principal concern is the fuel Peak Cladding Temperature (PCT).

- Challenges to the containment wherein the primary concerns are the maximum containment pressure calculated during the course of the LOCA and the maximum suppression pool temperature for long term cooling in accordance with 10CFR50, Appendix A, General Design Criterion (GDC) 38.

- The calculated radiological consequences of DBAs compared to the criteria of 10CFR100.

The ECCS are described in Section 6.3 of the LGS, Units 1 and 2 Updated Final Safety Analysis Report (UFSAR). As mentioned above, a complete spectrum of pipe breaks was investigated from the largest recirculation line down to instrument lines. The ECCS performance evaluation for power rerate was conducted through application of the 10CFR50, Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. Therefore, the ECCS safety margin is not impacted by power rerate.

Analyses provide the results of the plant containment response to various hypothetical LOCAs. For power rerate conditions, the containment pressure and bulk suppression pool temperature will slightly increase. However, containment pressures and temperatures following any DBA, have been evaluated to be acceptable. Thus, the containment and its cooling systems are judged to be satisfactory for rerated power operation.

The UFSAR provides the radiological consequences for each of the DBAs. These events have been re-evaluated using the methodology described in UFSAR Chapter 15 and the results remain below the 10CFR100 guideline values. Therefore, all radiological safety margins are maintained.

The effects of plant transients were evaluated by investigating a number of disturbances of process variables (e.g., temperature, pressure, flow), and malfunctions or failures of equipment according to a scheme of postulating initiating events. These events are primarily evaluated against the fuel Safety Limit Minimum Critical Power Ratio (MCPR). The Safety Limit MCPR is determined using NRC approved methods. The Operating Limit MCPR will be increased very slightly to assure that the Safety Limit MCPR is maintained, when a transient is initiated from the rerated power level. Therefore, the fuel margin of safety will not be affected by operating at power rerate conditions.

All of the other radiological releases previously evaluated are either unchanged because they are not power dependent, or increase by at most the percent increase in power. The dose consequences for all of the non-LOCA radiological release accident events are bounded by the radiological release events discussed above and remain within the regulatory limits for each release.

Plant equipment and instrumentation have been evaluated against the criteria in the current equipment Environmental Qualification (EQ) program at rerated conditions. The majority of equipment was determined to be qualified for the revised environmental conditions. Equipment that was determined to not remain qualified for the rerated conditions will be replaced prior to operation at the rerated power level.

Other systems and/or equipment used to perform safety-related and normal operating functions have been reviewed relative to rerated conditions in a manner comparable to that for safety-related Nuclear Steam Supply System (NSSS) systems and/or equipment. This includes, but is not necessarily limited to, all or portions of the following systems; Main Steam, Feedwater, Main Turbine, Condenser, Condensate, Essential and Non-essential Service Water, Ultimate Heat Sink (i.e., Spray Pond), Emergency Diesel Generators, Standby Gas Treatment, Spent Fuel Pool Cooling, and associated piping and supporting systems. Significant groups and/or types of equipment and/or systems were justified for rerated conditions by bounding evaluations. Unique evaluations justified power rerate operation for systems and/or equipment that were not generically justified.

Other special events and features such as Station Blackout, Fire Safe Shutdown, Fire Protection Features and Motor Operated Valves (MOVs) have been evaluated to ensure safe plant operation for rerated conditions.

The LGS Station Blackout analyses were reviewed to ensure that operation at rerated conditions does not compromise the commitments to Regulatory Guide 1.155, "Station Blackout," dated August 1988 and the compliance to 10CFR50.63. The effects of operation at rerated conditions were evaluated for areas required to mitigate the Station Blackout event. Pertinent equipment was shown to remain operable and have adequate capacity to mitigate this event ensuring safe shutdown under Station Blackout conditions.

The plant fire safe shutdown and fire protection features have been reviewed and there is no impact to the fire detection or suppression systems due to rerated conditions. The Fire Protection Evaluation Report (i.e., UFSAR Chapter 9, Appendix 9A) was reviewed to evaluate the impact of power rerate. ECCS and RCIC pump room heatup rates were examined and temperature responses were updated for rerate effects. Equipment was verified to be operational for these events and to have adequate capacity for rerated conditions.

All valves in the LGS MOV program (i.e., described in the response to NRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989) were evaluated. Valves which have been previously tested have been shown to be adequate for rerated conditions. The remaining few MOVs not yet tested under the MOV program will be reconciled or modified to be adequate for rerated conditions.

B. Information Supporting a Finding of No Significant Hazards Consideration

We have concluded that the proposed changes to the Limerick Generating Station (LGS), Units 1 and 2 Operating Licenses and their corresponding Appendices A (i.e., Technical Specifications (TS)), to reflect the planned implementation of the Power Rerate Program and the corresponding increase in the authorized maximum reactor core power level by 5%, do not constitute a significant hazards consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1) The proposed Operating License (OL) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed power rerate imposes only minor increases in the plant operating conditions. Plant systems, components, and structures have been verified to be capable of performing their intended functions under rerated conditions. Where necessary, some components will be modified or replaced prior to implementation of the Power Rerate Program to accommodate the revised operating conditions. No new component or system interactions that could lead to an accident are created. As discussed below, no transient events result in a new sequence of events which could lead to a new accident scenario.

   Emergency Core Cooling Systems (ECCS) - Loss-of-Coolant Accident (LOCA) Analysis

   The current ECCS-LOCA performance analysis is already bounding for power rerate conditions. The fuel peak cladding temperature for rerate conditions is 1,345°F, which is below the 2,200°F regulatory limit. Therefore, the analysis demonstrates that the LGS, Units 1 and 2 will continue to comply with 10CFR50.46 and 10CFR50, Appendix K.

   Transient Event Analysis

   The evaluation results for transient events indicate the margin to the fuel Safety Limit Minimum Critical Power Ratio (MCPR) will be maintained for the 8x8 array fuel types, such as GE8x8NB or GE11 fuel design. The current fuel thermal-mechanical limits will continue to be met.

   Also, the power-dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits developed as part of the Average Power Range Monitor Rod Block Monitor Technical Specifications (ARTS) improvement program are applicable to power rerate. A TS Change Request to implement the ARTS improvement program was submitted to the NRC by letter dated August 27, 1993. The peak reactor vessel bottom head pressure will remain within the American Society for Mechanical Engineers (ASME) Code requirement for reactor overpressure protection.

   The analysis performed focused on the most limiting transient events in each disturbance category selected specifically for the power rerate evaluations. The results demonstrated that LGS, Unit 1 and Unit 2 core thermal power output can be safely increased to power rerate parameters without impacting plant safety during a postulated transient event. The details of the impact to the description in the UFSAR are delineated below.

a) Events Resulting in a Core Coolant Temperature Decrease

i) Loss of Feedwater Heating (LFWH)

The delta Critical Power Ratio (ΔCPR) for the LFWH event at the rerated power is bounded by the result estimated for the current rated power level and remains significantly less than the Operating Limit MCPR. There is no change between the ΔCPR results for high and low reactor core flow conditions. The calculated thermal and mechanical overpowers for this event at power rerate conditions also meet the fuel design criteria.

ii) Feedwater Controller Failure (FWCF) Maximum Demand

For the Increased Core Flow (ICF) and the Maximum Extended Load Line Limit (MELLL) conditions, the trend for the FWCF - Maximum Demand event at rerate conditions is consistent with the current rated power analysis. For both high and low reactor core flow conditions, the FWCF - Maximum Demand event becomes most limiting due to the Turbine Bypass Valve Out-of-Service (TBVOOS) and the Recirculation Pump Trip Out-of-Service (RPTOOS) analyses assumption. The fuel thermal margin results remain within the acceptable limits for the fuel type analyzed.

b) Events Resulting in a Reactor Pressure Increase

i) Turbine Trip with No Bypass (TTNBP)

At rerate conditions, the fuel transient thermal and mechanical overpower results remain below the NRC acceptance criteria.

ii) Generator Load Rejection with No Bypass (LRNBP)

The fuel transient thermal responses are less severe than for the TTNBP event described above. Therefore, at power rerate conditions, the LRNBP event remains bounded by the TTNBP event.

iii) Main Steam Isolation Valve Closure, Flux Scram (MSIVF)

The peak reactor vessel bottom head pressure for rerate conditions is slightly higher than the pressure at current rated conditions due to the higher initial reactor coolant system pressure. However, this result is still below the ASME overpressure limit of 1,375 psig by a margin of 33 psi.

c) Events Resulting in a Core Coolant System Flow Rate Decrease

i) Recirculation Pump Seizure

The recirculation pump seizure assumes instantaneous stoppage of the pump motor shaft of one recirculation pump. As a result, the reactor core flow decreases rapidly. The reactor vessel level swell due to the rapid reactor core flow reduction reaches the high reactor water level setpoint, causing a feedwater pump trip, a main turbine trip, and subsequently a reactor scram on turbine stop valves closure. The peak neutron flux and average fuel surface heat flux do not increase significantly above the initial conditions, therefore no impact on the fuel thermal margin is postulated to occur.

d) Events Resulting in Reactivity and Power Distribution Anomalies

i) Rod Withdrawal Error (RWE)

The calculated ΔCPR of 0.10 for this event at rerate conditions is bounded by the generic ARTS - based RWE limits of 0.13. Therefore, the generic ARTS-based RWE analysis ΔCPR result is verified to be applicable for power rerate conditions for LGS Units 1 and 2.

e) Events Resulting in a Reactor Coolant Inventory Increase

i) Inadvertent High Pressure Coolant Injection (HPCI) System Actuation

Based on the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event (i.e., the TTNBP event described above) for ΔCPR consideration.

Anticipated Transients Without Scram (ATWS) Analysis

A generic evaluation for the ATWS event is provided in Section 3.7 of the Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Supplement 1, dated July 1991. This evaluation concludes that the ATWS acceptance criteria for fuel, reactor pressure vessel (RPV) and containment integrity will be met, if the following exists:

- Reactor power increases ≤ 5%;
- Reactor Steam Dome pressure increases ≤ 40 psi;
- Safety Relief Valve (SRV) opening setpoints increase ≤ 80 psi; and
- ATWS high pressure setpoint increases ≤ 20 psi.

The plant's parameter changes will remain within the above criteria, except that the ATWS high pressure setpoint increase is 40 psi rather than 20 psi in order to maintain the same relationship between the ATWS high pressure setpoint and the SRV opening setpoints. Based on the previous analysis, this difference would have a minor effect on the analysis results. The only significant change is a slightly higher (i.e., about 10 psi) peak RPV pressure.

For additional assurance, a LGS specific ATWS analysis for a 5% power rerate was performed. The events analyzed were:

1. Main Steam Isolation Valve (MSIV) Closure,
2. Pressure Regulator Failure - Open,
3. Loss of Feedwater, and
4. Inadvertent Opening of a Relief Valve.

The LGS specific analysis also concludes that the ATWS acceptance criteria for fuel, RPV, and containment integrity will be met for a 5% power rerate.

Other Evaluations

The impact of power rerate on the radiological consequences of the accidents presented in UFSAR Chapter 15 was determined based on the current design basis analyses, post rerate implementation system conditions, and radiological source terms. In general, power rerate will result in a small increase in the quantity of radioactive material released during accidents and therefore slightly higher (i.e., approximately 2% to 5%) accident doses. However, UFSAR Chapter 15 accident doses for rerated conditions remain within the regulatory limits specified in 10CFR100 and 10CFR50, Appendix A, GDC 19.

The UFSAR Chapter 15 accidents that were evaluated and updated for rerate conditions are as follows.

1) Loss of Coolant Accident (LOCA)
2) Main Steam Line Break (MSLB)
3) Fuel Handling Accident
4) Control Rod Drop Accident
5) Instrument Line Break
6) Feedwater Line Break
7) Steam Jet Air Ejector Line Break
8) Offgas System Failure
9) Liquid Radioactive Waste System Failure

An evaluation was also performed to address the power rerate impact on accident mitigative features, structures, systems, and components within the balance of plant. The results are as follows.

- Auxiliary systems such as the Emergency Service Water, Residual Heat Removal (RHR) Service Water, Ultimate Heat Sink (i.e., the spray pond), safety-related portions of secondary containment reactor enclosure air cooling, primary containment drywell air recirculation, and Emergency Diesel Generator enclosure ventilation were confirmed to operate acceptably under normal and accident conditions after implementation of power rerate.

- Combustible gas control systems were confirmed to be capable of maintaining oxygen concentrations inside the primary containment within regulatory limits under post accident rerate conditions.

- The secondary containment reactor enclosure recirculation system and Standby Gas Treatment system were confirmed to be able to adequately contain, process, and control the release of normal and post-accident levels of radioactive material after implementation of power rerate.

- Instrumentation was reviewed and confirmed to be capable of performing their control and monitoring functions under rerate conditions.

- Electric power systems including the main turbine generator and switchgear components were verified as being capable of providing the electrical load as a result of the rerated power levels. No safety-related electrical loads were affected which would impact the Emergency Diesel Generators.

- Piping systems were evaluated for the effect of operation at higher power levels, including transient loadings. The evaluation confirmed that with few exceptions piping and supports are adequate to accommodate the increased loadings resulting from operation at rerated power conditions. In a few cases, piping supports will be modified to accept the higher forces due to rerate conditions.

- The effect of rerate conditions on high energy line break (HELB) events for all Nuclear Steam Supply System (NSSS) and Balance of Plant (BOP) systems was evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement and blowdown forces and the environmental effects resulting from HELB events at rerate conditions.

- The Moderate Energy Line Break (MELB) analysis was evaluated for impact due to rerate conditions. Sufficient margin was determined to exist in the original analysis to bound the rerate conditions.

- Main control room (MCR) habitability was evaluated. Post-accident MCR and Technical Support Center (TSC) doses were confirmed to be within the limits of General Design Criterion (GDC) 19 of 10CFR50 Appendix A.

- Radiation doses for normal operation were reviewed and confirmed to remain within the limits of 10CFR20 and 10CFR50, Appendix I. The impact on post-accident sampling activities and post-accident access to vital areas was also confirmed to be acceptable.

- The environmental qualification of electrical and mechanical equipment important to safety was evaluated for the impact of normal and accident operating conditions at rerated power levels. The majority of equipment will remain qualified for the new conditions. For equipment that is not qualified, corrective actions will be taken to ensure the plant equipment will perform their intended functions under rerate conditions. No new equipment will be added for power rerate which would increase the potential for component failure. The Preventative Maintenance Program (PMP) will continue to provide for appropriate equipment repair or replacement during operation at rerated power conditions.

- The impact of operation at rerated power levels was evaluated for Station Blackout and Fire Safe Shutdown area heat-up concerns. The evaluation confirmed there is no adverse impact from rerate on the ability of the plant to achieve safe shutdown under these conditions.

The consequences of postulated transients and special events (i.e., ATWS and Station Blackout) will remain within NRC acceptance criteria for rerate conditions. Concurrent malfunctions assumed to occur during accidents have been accounted for in the safety analyses for rerate conditions. The consequences of these equipment malfunctions will not change with implementation of the Power Rerate Program. Equipment that is important to safety either is capable of or will be modified and/or replaced to be capable of performing its intended function. The availability of redundant systems to provide safety functions in the event of component malfunction is not impacted as a result of rerate conditions. Furthermore, the impact of power rerate on the consequences of abnormal transients and accident conditions which are a result of component malfunctions has been shown to be acceptable.

The probability (i.e., frequency of occurrence) of Design Basis Accidents (DBAs) occurring is not affected by the proposed increased power level, as the applicable regulatory criteria established for plant equipment (e.g., ASME Code, the Institute of Electrical and Electronics Engineers (IEEE) standards, National Electrical Manufacturer's Association (NEMA) standards, NRC Regulatory Guides) will still be followed as the plant is operated at the rerated power level. Reactor SCRAM setpoints will be established such that there is no significant increase in frequency due to rerate conditions. No new challenges to safety-related equipment will result from the implementation of power rerate.

The changes in consequences of hypothetical accidents which would occur from 102% of the rerated power, compared to those previously evaluated, are in all cases not significant, because the accident evaluations from a power rerate to 105% of original rated power will not result in exceeding the applicable NRC approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and has been determined to meet the current regulatory criteria for LGS, Units 1 and 2 at rerate conditions. The offsite radiological doses resulting from DBAs are calculated to increase by only a few percent (i.e., approximately 2% to 5%) because of the rerated power level, and will remain below 10CFR100 limits. In the area of reactor core design, the fuel operating limits will continue to be met at the rerated power level, and fuel reload analyses will continue to show that plant transients will meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II."

Challenges to fuel or ECCS performance were evaluated and shown to still meet the criteria of 10CFR50.46 and 10CFR50, Appendix K. Challenges to the primary containment have been evaluated and still meet 10CFR50, Appendix A, GDC 38, "Long Term Cooling," and GDC 50, "Containment." Radiological release events have been evaluated and have been shown to meet the guidelines of 10CFR100.

Therefore, the proposed OL changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) The proposed OL changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

All actions to ensure that safety-related structures, systems, and components will remain within their design allowable values, and ensure that they can perform their intended functions under rerate conditions will be taken prior to implementation of power rerate. Power rerate does not increase challenges to or create any new challenges to safety-related equipment or other equipment whose failure could cause an accident. No new equipment is added as a result of implementing the Power Rerate Program which would create the possibility of a new type of accident. In addition, power rerate does not create any new sequence of events or failure modes that lead to a new type of accident.

Implementation of power rerate will increase the average neutron flux in the reactor core, which increases the integrated neutron fluence on the reactor pressure vessel (RPV) wall. To account for the higher fluence, a RPV fracture toughness analysis was performed for power rerate conditions. This analysis resulted in a proposed revision to the "pressure vs. temperature" curves currently provided in the Technical Specifications (TS), that will maintain the current level of protection for the RPV. Therefore, power rerate will not result in any new failure mode for the RPV, and thus, does not create the possibility of a different type of accident from any accident previously evaluated.

No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as resulting from the implementation of the Power Rerate Program. The full spectrum of accident considerations defined in NRC Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," Revision 3, dated November 1978, have been evaluated for rerate conditions and no new or different kind of accident has been identified. Implementation of the Power Rerate Program uses already-developed technology and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria to include applicable NRC approved codes, standards, and methods. General Electric (GE) has designed Boiling Water Reactors (BWRs) of higher power levels than the rerated power of any of the currently operating BWR fleet and no new power dependent accidents have been identified.

Therefore, the proposed OL changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) The proposed OL changes do not involve a significant reduction in a margin of safety.

Power rerate will not involve a significant reduction in a margin of safety, as plant equipment and reactions to transients and hypothetical accidents will not result in exceeding the presently approved NRC acceptance limits. The accident doses are calculated to increase a few percent (approximately 2% to 5%) because of power rerate, but remain below 10 CFR 100 limits.

The events (i.e., transients, accidents, and ATWS) that form the bases of the TS were evaluated for power rerate conditions. Although some changes to the TS are required to implement power rerate, no NRC acceptance limit will be exceeded. Therefore, the margins of safety with respect to the safety limits and other TS bases will be maintained.

For systems addressed in the TS Sections 2.2, 3/4.1, 3/4.2, 3/4.3, 3/4.4, 3/4.5, 3/4.6 and 3/4.7 (i.e., Reactor Protection System, Standby Liquid Control System, Power Distribution Limits, Instrumentation, Reactor Coolant System, Emergency Core Cooling Systems, Containment Systems, and Plant Systems), all components will be operable and capable of performing their intended functions under power rerate conditions such that the margin of safety is not adversely impacted.

Therefore, the proposed OL changes do not involve a significant reduction in a margin of safety.

III. Information Supporting an Environmental Assessment

An environmental assessment is not required for the changes proposed by this Operating License (OL) Change Request because the requested changes conform to the criteria for "actions eligible for categorical exclusion," as specified in 10CFR51.22(c)(9). The requested changes will have no impact on the environment.

The cooling tower increased heat load was reviewed for a possible impact on the Schuylkill River as well as water diversion limits from the Delaware River. The quantity and type of corrosion inhibitors and biocides which are currently used in the circulating water loops were reviewed to determine if they were impacted by implementation of rerate.

Increased heat loads on the cooling towers will slightly alter cooling tower blowdown operations and the use of water chemistry chemicals. The change in usage of chemicals will not impact existing National Pollutant Discharge Elimination System (NPDES) permit limitations. Impacts to air, water, and land resources will be non-existent.

Other non-radiological environmental impacts of the proposed power rerate were reviewed based on the information submitted in the Environmental Report, Operating License Stage, the NRC Final Environmental Statement (FES), Operating License Appendix B (i.e., Environmental Protection Plan), the requirements of the applicable NPDES permits, which includes the outfall limits, and the Delaware River Basin Commission (DRBC) Water Use permit. We have concluded the proposed power rerate will have insignificant impacts on the non-radiological elements of concern and the plant will be operated in an environmentally acceptable manner as established by the FES. Existing Federal, State and Local regulatory permits presently in effect will accommodate power rerate without modification.

The proposed changes do not involve a significant hazards consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

IV. Conclusion

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the Limerick Generating Station (LGS), Units 1 and 2 Operating License Nos. NPF-39 and NPF-85 and their corresponding Appendices A and have concluded that they do involve an unreviewed safety question, but that they do not involve a significant hazards consideration, and will not endanger the health and safety of the public.