Station Support Department

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PECO ENERGY

PECO Energy Company
Nuclear Group Headquarters
965 Chesterbrook Boulevard
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October 19, 1994

Docket Nos. 50-352
50-353

License Nos. NPF-39
NPF-85

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

SUBJECT: Limerick Generating Station, Units 1 and 2
Response to Request for Additional Information Regarding Power Rerate Program dated September 21, 1994 (RAI-3) and Submittal of Errata Sheet for NEDC-32225P

Gentlemen:

Attached is our response to your Request for Additional Information (RAI), dated September 21, 1994 regarding the planned implementation of the Power Rerate Program at Limerick Generating Station (LGS), Units 1 and 2, (Attachment No. 1).

The Power Rerate Program is the subject of Operating License Change Request No. 93-24-0 which was forwarded to you by letter dated December 9, 1993. Included in that package was the General Electric report NEDC-32225P, "Power Rerate Safety Analysis Report for LGS Units 1 & 2," dated September 1993. Table 5-1 of this report, "Analytical Limits for Setpoints," did not reflect the revised allowable value of 137°F for the Reactor Water Cleanup (RWCU) Heat Exchanger Room Temperature - High, at power rerate conditions. Therefore, we are submitting the Errata and Addenda Sheet No. 1 for NEDC-32225P, providing this correction (Attachment No. 2). The proposed Technical Specifications (TS) changes (i.e., TS page 3/4 3-19 for LGS Units 1 and 2), forwarded to you by our December 9, 1993 letter, reflect the correct allowable value of 137°F for the RWCU Heat Exchanger Room Temperature - High, at power rerate conditions and are not affected by the Errata Sheet provided in Attachment No. 2.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

G. A. Hunger, Jr.

G. A. Hunger, Jr.
Director - Licensing

Attachments

cc: T. T. Martin, Administrator, Region I, USNRC - w/ attachments
N. S. Perry, USNRC Senior Resident Inspector, LGS - w/attachments
R. R. Janati, Director, PA Bureau of Radiological Protection - w/attachments

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COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF CHESTER :

W. H. Smith, III, being first duly sworn, deposes and says:

That he is Vice President of PECO Energy Company, the Applicant herein; that he has read the enclosed information concerning Operating License Change Request No. 93-24-0 (Power Rerate Program) for Limerick Generating Station Facility Operating License Nos. NPF-39 and NPF-85, 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.

William H Smith III

Vice President

Subscribed and sworn to

before me this 19th day

of October 1994.

Erica A. Santori

Notary Public

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

Docket Nos. 50-352
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ATTACHMENT NO. 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI-3)
LIMERICK GENERATING STATION, UNITS 1 AND 2

OPERATING LICENSE CHANGE REQUEST NO. 93-24-0

Reference: "Power Rerate Safety Analysis Report For Limerick Generating Station, Units 1 & 2," General Electric Company, NEDC-32225P, Class III, September 1993 (proprietary)

Question 1:

The evaluation (Section 2.5.1) did not address the effects of the power uprate such as the increase of the reactor pressure and temperature, and the high pressure scram setpoint on the structural and functional integrity of the control rod drive system (CRDS). State the basis for determining the acceptability of the CRDS regarding compliance with the design Code. The information provided should include the Code and Edition, the Code allowables, the calculated maximum stresses, deformation, and fatigue usage factor for the uprated power conditions, and assumptions used in the calculations.

Response 1:

The Limerick Generating Station (LGS) Units 1 and 2 CRDS were evaluated for a 1045 psig reactor dome pressure and an additional 35 psid for the vessel bottom head. The CRD mechanism structural and functional integrity was deemed acceptable for the vessel bottom head pressure of 1080 psig.

The components of the CRD mechanism designated as primary pressure boundary have been designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. The applicable ASME Code effective date for the initially supplied Limerick CRDS is the 1968 Edition up to and including Winter 1969 addenda. The limiting component of the CRD mechanism is the indicator tube which has a calculated stress of 20,790 psi (allowable is 26,060 psi). The maximum stress is due to a maximum CRD internal hydraulic pressure of 1750 psig. The analysis for cyclic operation of the CRD was conservatively evaluated in accordance with ASME Code, N-415.1 (NB-3222.4(d)). All requirements of N-415.1 (NB-3222.4(d)) are satisfied even when considering the increased vessel bottom head pressure at rerate conditions, thereby satisfying the peak stress intensity limits

governed by fatigue. It should be noted that the CRDS has been successfully tested for all operational modes at simulated reactor vessel pressures up to 1250 psig saturated conditions, which bounds the rerated high pressure scram setpoint pressure of 1111 psig. Additional analysis shows that the maximum usage factor calculated based on NB-3222.4(d) is 0.15 for the CRD main flange, which is less than the allowable limit of 1. Based on the adequate stress margins and successful testing, it is concluded that the deformations are reasonable, satisfactory, and not a concern at power rerate conditions.

For CRD insertion and withdrawal, the minimum specified differential pressure (250 psid) between the hydraulic control unit and the reactor vessel was evaluated for LGS power rerate conditions. Based on evaluation of plant CRD pump and system data, Units 1 and 2 CRD pumps will be modified to provide adequate pump head and flow for CRD positioning and cooling at power rerate conditions. During power operation, the primary scram pressure for the CRDs is provided by the pressure inside the reactor vessel. Therefore, the CRDS will perform all its safety functions at rerate conditions at LGS Units 1 and 2.

Question 2:

Discuss the planned CRD pump modifications with regard to the specific pump changes, quantify adequate pump head and flow that will be provided, modification schedule and operational experience of CRD at similar operating conditions. (Reference Section 2.5.1)

Response 2:

A modification has been initiated at LGS to replace the existing pump, motor and gear box with new higher capacity direct drive (i.e., pump and motor, no gear box) pumps. The modification will include minor piping configuration changes to support pump fitup. Also, electrical setpoints associated with the new motors will be incorporated.

This modification is being performed because, as a result of the evaluation for power rerate, it was determined that the existing pumps do not have sufficient capacity under rerate conditions during normal CRD positioning operations. This is due to high line losses from the discharge of the pump to the CRD flow control station. Maintaining the required differential pressure of 250 psid between the CRD system and reactor pressure would result in cooling water flows to decrease to approximately 35 gpm. Subsequently, this may result in the increase of the number of drives running hot. After discussion with the existing pump manufacturer and review of the pump operating history, it was determined that the pumps should be replaced. This replacement includes pumps and motors and the removal of the existing pumps, motors, and gear boxes.

The new pumps are designed to maintain pre- and post- rerate condition CRDS flow of 93 gpm. However, the required pump discharge pressure will be increased from 1465 psig to 1550 psig. This increase in design discharge pressure includes adjustments for rerate conditions and the high line losses described above.

The modification is planned for November and December 1994 for LGS Unit 2, and Spring of 1995, for LGS Unit 1.

The modification will not affect the design flow rate for the CRD system, only the pump discharge pressure. The discharge pressure is within the design pressure of the CRD piping. Due to the high line losses between the CRD pumps and the flow control station, pressure at the flow control station and downstream of it should be similar to that which is currently seen prior to power rerate.

Question 3:

Section 3.2 indicates a rerated high peak reactor pressure vessel (RPV) bottom head pressure of 1342 psig, which is below the ASME limit of 1375 psig. Provide information on the high peak RPV bottom head pressure without the rerated conditions, and an explanation of the assertion that there is no decrease in margin of safety relative to the rerated pressure.

Response 3:

The limiting pressurization event (i.e., the Main Steam Isolation Valves (MSIV) closure with a failure of valve position scram) was analyzed at 102% of current power, 105% of rated core flow and with all safety relief valves (SRVs) in-service. The calculated peak RPV bottom head pressure was 1260 psig. For power rerate, this case was analyzed with 102% of rerated power, 110% of rerated core flow, with all SRVs in-service and with three SRVs out-of-service as well. The calculated peak RVP bottom head pressures were 1314 psig and 1342 psig, respectively.

In accordance with General Electric (GE) report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Volume 1, Licensing Topical Report, Class III, July 1991, Section 5.0, the safety margin prescribed by the Code of Federal Regulations (CFR), as well as the margin specified by the ASME Code will be maintained. The margin of safety is inherent in the 1375 psig ASME limit.

The overpressure analysis for LGS Units 1 and 2 power rerate conditions demonstrates that the peak bottom pressure remains below the 1375 psig ASME limit; therefore, there is no decrease in the margin of safety.

Question 4:

Specify "other dynamic loads" included in the reactor internals load combinations. Provide a discussion on the effects of power uprate on the reactor internals responses associated with loss-of-coolant-accident (LOCA), safety/relief valve discharge, annulus pressurization and jet reaction loads. (Reference Section 3.3.2)

Response 4:

In addition to seismic loads, the LGS Units 1 and 2, reactor vessel and internals design considered a number of Mark II Containment related dynamic loads, often referred to as "new loads," in NEDC-30493 (Unit 1, 1984) and NEDC-31670 (Unit 2, 1989) (an evaluation was performed after the original design effort). Dynamic loads on the vessel and internals are generally determined by dynamic analyses using a lumped mass model of the vessel and internals. Specific phenomena considered in these "new loads" and appropriate references are summarized in Table 1 below. These are the "other dynamic loads" referred to in Section 3.3.2.

Table 1: Other Dynamic Loads for Reactor Vessel and Internals

Load Designator (see table 201 NEDC-30493)	Description of Load Source	Specific References (see Appendix A NEDC-30493)
SRV	One valve actuating	NSSS SRV Load Evaluation for LGS VPF 299X126-049 (Nuclear Services Inc.)
SRV_{all}	Actuation of all valves within milliseconds of each other	NSSS SRV Load Evaluation for LGS VPF 299X126-049
SRV_{ads}	ADS operation	NSSS SRV Load Evaluation for LGS VPF 299X126-049
VC	Vent clearing	22A7682 Dynamic Loads Report-Loss of Coolant Accident
AP	Annulus pressurization	NSSS Annulus Pressurization Load Evaluation for LGS VPF 299X126-049 (Nuclear Services Inc.)
CHG	Chugging	22A7682 Dynamic Loads Report-Loss of Coolant Accident
FLL	Vertical Load from Fuel Lift (Static Equivalent)	22A7683 Rev 0, Dynamic Loads Report - Fuel Lift

The effect of power rerate on these dynamic loads was investigated, and it was concluded that only the fuel lift load for the top guide was impacted. The small increase, less than 10%, was considered in the evaluation of the top guide. All appropriate combinations of the other dynamic loads were considered along with weight, pressure differential and seismic for the reactor internals evaluation.

Question 5:

The evaluation of reactor internals in Section 3.3.2 did not address the Code and Edition used for evaluating stresses and allowables for the reactor vessel and internals. Provide such information and list the maximum stresses, fatigue usage factor and location of highest stress areas for both the current design and the uprated power conditions.

Response 5:

The evaluation of the reactor internals uses ASME Boiler and Pressure Vessel Code, Section III, as a guide for design acceptance criteria. The specific applicable Code Edition for the reactor pressure vessel, including the shroud support is the 1968 Edition with addenda up to and including the Winter 1969 addenda.

The stresses or loads for 18 reactor internal components were evaluated by either confirming that rerate load combinations are bounded by previous analyses or scaling stresses using conservative load ratios from these analyses. In some cases, previous analyses were repeated as required to demonstrate acceptance. The structural integrity of the component is demonstrated by comparison with applicable allowable stresses. The stress results for three of the highest stress components are shown in Table 1 below. For power rerate conditions, the reactor pressure vessel closure head studs had the highest reactor internal usage factor, calculated to be 0.95.

TABLE 1

COMPARISON OF MAXIMUM STRESSES AND THEIR LOCATIONS IN LIMERICK 1 AND 2 REACTOR INTERNALS AT CURRENT AND POWER RERATE CONDITIONS

UPSET CONDITIONS

Component	Maximum Stress[1] Location	Maximum Stress Comparison[2] Current	Maximum Stress Comparison[2] Rerated
Steam Dryer	Bank attachment	NA[3]	27.9 ksi (28.0 ksi)
Shroud Support	Support plate	33.64 ksi (34.95 ksi)	Loads less than original RPV stress report loads
Shroud Head	Lugs	13.8 ksi (25.4 ksi)	18.0 ksi (25.4 ksi)

FAULTED CONDITIONS

Component	Maximum Stress[1] Location	Maximum Stress Comparison[2] Current	Maximum Stress Comparison[2] Rerated
Steam Dryer	Bank Attachment	60.3 ksi (60.8.ksi)	No affect
Shroud Support	Leg	40.4 ksi (42.2 ksi)	41.3 ksi (42.2 ksi)
Shroud Head	Lugs	27.9 ksi (60.8 ksi)	Greater margin than upset, not specifically evaluated

1. Maximum stress locations are locations where the margin to the allowables is the smallest.
2. Allowable values are shown in parentheses.
3. Previous evaluations did not consider upset condition applicable to the steam dryer.

Question 6:

Section 3.3.3.2 states, "Elastic-plastic methods were implemented for some components; the Code requirements for these methods were met." Provide a detailed discussion on the analysis methodology, assumptions and compliance with the Code including Edition, and the Code allowables used, with regards to acceptability of stress levels and fatigue considerations.

Response 6:

According to Sections NB-3222 and NB-3223 of the ASME Code, structural adequacy is met if the maximum primary plus secondary stress intensity range (S_n) at a location on the component is less than 3 S_m of the material. If the 3 S_m limit is not met, then plastic behavior is assumed and the simplified elastic-plastic analysis of the ASME Code, Paragraph NB-3228.3, can be used to determine structural adequacy.

The 3 S_m limit on the range of primary plus secondary stress intensity may be exceeded provided the following requirements are met.

a. The range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses, shall be $\leq 3S_m$.

b. The value of S_a used for entering the design fatigue curve is multiplied by the factor K_e.

c. The rest of the fatigue evaluation stays the same.

d. The component meets the thermal ratcheting requirement.

e. The material temperature does not exceed the maximum temperature permitted for the material.

f. The material shall have a specified minimum yield strength to specified minimum tensile strength ratio of less than 0.80.

The power rerate stress analysis uses the guidelines and procedures of the ASME Boiler and Pressure Vessel Code, Section III. For the component under consideration, the 1968 ASME Code, "Rules for Construction of Nuclear Vessels," ASME Boiler and Pressure Vessel Code Section III, 1968 Edition, which is the Code of construction shall be the governing Code. However, if a component underwent a design modification, the governing Code shall be the Code used in the stress analysis of the modified component. Also, if the original analysis used a later version of the Code, that later version is applicable. In some cases, the 1989 version of the ASME Code "Rules for Construction of Nuclear Vessels," ASME Boiler and Pressure Vessel Code Section III, 1989 Edition was used to determine material properties.

Question 7:

Table 3-4 provides fatigue usage factors of the limiting components of the reactor vessel and its support. Notes (1) through (3) of this table imply that the maximum fatigue usage life for the feedwater nozzle is reduced from the current 32 years to 23 years for the power uprate. Provide a discussion of the program for monitoring, planned refurbishment and schedule.

Response 7:

The surveillance test for thermal transient monitoring has a provision for evaluating the actual service life of the thermal sleeve with respect to its allowable life. This procedure requires that the refurbishment process be initiated within 4 years of the allowable service life, and will be revised to reflect the new feedwater nozzle fatigue usage life of 23 years.

Question 8:

Section 3.5 indicated that "The evaluation of all potentially affected systems, except for the main steam inside containment and recirculation piping, are described in Section 3.12." However, Section 3.12 evaluated only the feedwater and BOP piping. List all NSSS piping systems (large and small bore piping of the reactor pressure boundary) that were evaluated for the uprated power conditions. Provide a detailed discussion on the evaluation of these piping systems, components (pumps, valves, nozzles , penetration, etc.) and supports (hangers, snubbers and struts including anchorage, thermal and vibration displacements), and compliance with the Code including Edition, and the Code allowables used, with regards to acceptability of stress levels and fatigue considerations.

Response 8:

As stated in Section 3.5 of NEDC-32225P all potentially affected piping systems (including NSSS piping), except for main steam inside containment and recirculation piping are described in Section 3.12. Main steam (inside containment) and recirculation system evaluations are described in Section 3.11.

Despite the title "Feedwater and BOP Piping", Section 3.12 provides pipe stress and support evaluation methodology and results as described in Sections 3.12.1, 3.12.2, and Table 3-5. Section 3.11 provides the evaluation methodology and results for main steam piping (inside containment) and recirculation systems. No piping or support modifications are required for operation at rerate conditions due to initial evaluations performed or subsequent reconciliations performed within the limits of applicable code requirements and Updated Final Safety Analysis Report (UFSAR) commitments.

Question 9:

Section 3.12 states that "In the cases where unacceptable results were encountered, minor modifications or refined analyses will be made ..." Specify piping systems and supports that require modification for the power uprate. Provide a discussion on modifications and the analytical method and assumptions of refined analyses to be performed for the unacceptable cases with the proposed power uprate conditions.

Response 9:

Minor piping and support modifications anticipated in Section 3.12 at the time of the original submittal have been reconciled, and it was concluded that no piping and support modifications are required for operation at rerate conditions. Reconciliation, where required, was based on removal of conservatism in the existing calculations permitted within the limits of applicable code requirements and existing UFSAR commitments.

Question 10:

State the Code and Edition used for the power uprate evaluation of balance-of-plant (BOP) piping, in-line components (valves and nozzles, etc.), and pipe supports including anchorages. List the limiting BOP piping systems and components with respect to the maximum stresses and safety margin as a result of the power uprate. (Reference Sections 3.12)

Response 10:

The Code and Edition used for the power rerate evaluation of all balance-of-plant (BOP) piping, in line components and pipe supports including anchorages is the same as that committed to in LGS UFSAR (Table 3.2-1 and other various subsections of Chapter 3).

The acceptability of operation at the rerated conditions of the piping and components in all affected plant systems (excluding main steam and recirculation systems) is addressed in Section 3.12, "Feedwater and BOP piping" of NEDC-32225P. The limiting balance of plant systems are the main steam relief valve discharge, main steam outside containment, and feedwater systems.

In most cases, calculated pre-rerate stresses were at least 10% below code allowable. The maximum stress levels and fatigue analysis results for all BOP piping were reviewed based on bounding increases in temperature, pressure and flow rate and are

shown in Table 3-5 of NEDC-32225P. In general, rerate conditions account for a small (less than 5%) increase in stress levels over current conditions. In a few cases, the pre-rerate stresses were within 5% of code allowable. However, this piping was evaluated and determined to be acceptable, since the stresses were still below code allowable at rerate conditions.

Auxiliary systems (i.e., BOP), impacted by power rerate have been evaluated and determined to meet the acceptance criteria of the code of record for operation at rerate conditions.

Question 11:

Provide a discussion on the methodologies and assumptions used in the evaluation regarding the effects of the power uprate on the design basis analyses of the high energy line break locations, and pipe-whip and jet impingement loads. (Reference Sections 3.12.1, 10.1 and 10.1.1)

Response 11:

The piping system evaluations were based on process system conditions at the rerated power level. Specific increases in temperature, pressure, and flow rate for each evaluated system were considered. No new assumptions, other than those identified in LGS UFSAR were considered. The effect of power rerate on pipe break postulation and location was evaluated. It was determined that sufficient margin exists to accommodate rerate with no additional break locations required. Effects of power rerate conditions on pipe whip and jet impingement loads are discussed in Section 10.1.2 of NEDC-32225P. Based on the review completed since the original power rerate submittal, it is determined that the existing structures and structural components are adequate to support loads at rerated conditions.

Question 12:

Section 10.1.1 states that "in a few cases where unacceptable results were encountered, structural elements and components will be qualified for the rerated HELB conditions by refined analysis, modification or replacement prior to rerate implementation." Specify piping systems, components, pipe supports and postulated break locations that require modification and the analytical methods and assumptions of refined analyses to be performed for the unacceptable cases with the uprated power conditions.

Response 12:

Minor modifications anticipated for structural elements in Section 10.1.1 at the time of the original power rerate submittal have been reconciled, and it is concluded that no structural modification is required for operation at rerate conditions. Reconciliation, where required, was based on removal of conservatism in the existing calculations.

Question 13:

A recent abnormal reactor recirculation pump vibration issue has been reported by the Susquehanna licensee during testing for the power uprate conditions. Provide an evaluation of the increased flow-induced dynamic loads on the recirculation piping and components such as pumps and flow control valves, to assure that excessive recirculation pump vibration will not occur at Limerick for the power uprate conditions.

Response 13:

In accordance with GE Licensing Topical Report (LTR) 1, NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," dated May 1992, a review of LGS plant-specific data was performed to confirm that sufficient margin exists to accommodate minor changes in vibration displacements due to the change in recirculation pump speed (i.e., less than 1% of rated speed), required to provide the same core flow at the rerated conditions. Steady-state vibration and thermal displacement measurements obtained during the initial preoperation and startup of the plant were assessed for the impact of power rerate conditions. Existence of large margin to the conservatively set criteria were confirmed.

The LGS Power Rerate submittal does not request a change in the current maximum core flow limit (i.e., 105% core flow). An increase in core flow is not required to achieve power rerate conditions. In fact, due to the recent implementation of Average Power Range Monitor-Rod Block Monitor Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) at LGS, core flow will be less than rated flow for the majority of the operating cycle, including future cycles at rerated power. Under a separate program, PECO Energy has implemented 110% increased core flow prior to power rerate. This program is part of the long term strategy to improve fuel cycle costs. Under this program, PECO Energy has monitored the recirculation system and other related components to detect any signs of abnormal vibrations as the core flow was increased. 110% increased core flow operation has been implemented successfully at LGS Unit 2 at the current (pre-rerate) power level, and no abnormal vibration has been detected. Based on the discussions above, PECO Energy does not expect to experience any significant change in the recirculation system performance when the LGS Units 1 and 2 are rerated.

ATTACHMENT NO. 2

Errata and Addenda Sheet No. 1 for NEDC-32225P,
"Power Rerate Safety Analysis Report for
Limerick Generating Station Units 1 & 2,"
Class III, dated September, 1993