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ComEd

March 24, 2000

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to Request for Additional Information License Amendment Request for Power Uprate Operation

References: (1) Letter from R. M. Krich (Commonwealth Edison (ComEd) Company) to U.S. NRC, "Request for License Amendment for Power Uprate Operation," dated July 14, 1999.

(2) Letter from D. M. Skay (U.S. NRC) to ComEd, "Request for Additional Information – LaSalle County Station, Units 1 and 2 (TAC Nos. MA6070 and MA6071)," dated March 14, 2000.

In the Reference 1 letter, pursuant to 10 CFR 50.90, "Application for Amendment of License or Construction Permit," we proposed to operate both LaSalle County Station Units at an "uprate" power level of 3489 Megawatts Thermal (MWT). Per Reference 2, the NRC requested additional information concerning the proposed amendment request to support their review. The attachment to this letter provides our response to the request for additional information.

A001

The no significant hazards consideration, submitted in Reference 1, remains valid for the information attached.

Should you have any questions concerning this letter, please contact Mr. Frank A. Spangenberg, III, Regulatory Assurance Manager, at (815) 357-6761, extension 2383.

Respectfully,

Charles G. Pardee
Site Vice President
LaSalle County Station

Attachment

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – LaSalle County Station

STATE OF ILLINOIS	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON COMPANY	)	
LASALLE COUNTY STATION - UNIT 1 & UNIT 2	)	Docket Nos. 50-373 50-374

Subject: Response to Request for Additional Information License Amendment Request for Power Uprate Operation

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

for

Charles G. Pardee
Site Vice President
LaSalle County Station

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 24th day of March, 2000.
My Commission expires on 10-1, 2000.

OFFICIAL SEAL
DEBRA J. FEENEY
PUBLIC, STATE OF ILLINOIS
COMMISSION EXPIRES10-1-2000

Debra J. Feeney
Notary Public

Attachment
Response to Request for Additional Information

Question 1:

What design bases parameters, assumptions or methodologies (other than the source term) were changed in the radiological design basis accident analyses as a result of the proposed change? If there are many changes it would be helpful to compare and contrast them in a table. Also, please provide justification for any changes.

Response 1:

The power level and effective full power days (EFPDs) utilized for the source term development are identified in response to Question 2. For all other parameters (except the source term), the modeling is the same as described in the Updated Final Safety Analysis Report (UFSAR).

The modeling for the Main Steam Line Break (MSLB), Inadvertent Main Steam Isolation Valve (MSIV) closure, Instrument Line Break (inside secondary containment), Feedwater Line Break (FWLB), Radioactive Gas Waste System Leak or Failure, and Postulated Radioactive Releases due to Liquid Radwaste Tank Failure, are described in our responses to Questions 3 and 4 below.

Question 2:

NEDC-32701P, Revision 2, Section 8.3.2, states the following: "Consequently, in order to assure that the inventories of the long-lived isotopes are bounded, a limiting fission product activity inventory has been assessed based on a core irradiation of 1300 effective-full-power-days (EFPD) at a bounding power level." Please define the bounding power level for this analysis.

Response 2:

The power level and EFPD used in the power uprate evaluations are:

Accident/Evaluation	Power Level	EFPD
Loss-Of-Coolant	3910 MWt	2034 days
Fuel Handling	3559 MWt	2034 days and 1.0 day of decay
Control Rod Drop	3559 MWt	2034 days
Cask Drop	3489 MWt	1953 days and 360 days of decay
Environmental Qualification Evaluations	3908 MWt	1300 days (includes 10% IEEE margin)

These parameters bound the power uprate case of 3489 MWt for 1300 effective full power days and thus represent conservative bounding values for fission product inventory.

Question 3:

NEDC-32701P, Revision 2, Section 9.2, states the following: "The Main Steam Line Break Accident (MSLBA) outside containment for power uprate will release less steam and reactor coolant mass than the design basis modeling in the current Updated Final Safety Analysis Report (UFSAR) evaluation." This assessment of the MSLBA outside of containment does not address any potential increased source term, nor does it provide an assessment of the impact of the proposed change on this analysis. Please provide more justification and the resulting doses due to the proposed change.

Response 3:

For 105% power uprate, it was determined that the MSLB accident will release 100% of the steam and 98.8% of the reactor coolant when compared to the current UFSAR basis. The MSIV closure time is not changed by power uprate. The design basis coolant activity (which is based on Technical Specification 3/4.4.5 limits) is not impacted by power uprate. Therefore, the present UFSAR dose for the MSLB remains limiting.

Question 4:

NEDC-32701P, Revision 2, Section 9.2, discusses five "other" Chapter 15 events: Inadvertent main steam isolation valve (MSIV) Closure, Instrument Line Break, Feedwater Line Break, Radioactive Gas Waste System Leak or Failure, and Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure. This section provides an "expected" increase for each of the accidents, but does not provide a bases for these numbers. Please provide a justification for the "expected" increases and the dose results from these analyses.

Response 4:

NEDC-32701P, Revision 2, Section 9.2, "Design Basis Accidents," indicates that the potential anticipated change to any of the radiological doses is not expected to increase by more than 25%, with the exception of the feedwater line break. This estimate was a conservative, bounding estimate, and has been quantified to respond to the above question. The expected increase, if any, and the dose results from each of the above analyses are described individually below.

- Inadvertent MSIV Closure (UFSAR 15.2.4)

The model described in the current UFSAR Section 15.2.4 applies to the present evaluation of offsite dose impact for the inadvertent MSIV closure. There is no additional impact due to the 105% power uprate on the reactor coolant system, the steam line relief system, the primary containment structures, suppression pool structures, or offsite doses (which are controlled by station administration and procedures). An analysis demonstrates that fuel thermal margin requirements are met. Therefore, Inadvertent MSIV Closure at 105% power uprate does not result in fuel damage. Additionally, the reactor dome pressure and the design basis normal reactor coolant fission product inventory do not change. Therefore, the mass blowdown to the suppression pool and design fission product inventory in the suppression pool do not change for 105% power uprate. The radioactive material releases to the environment (containment purging) and primary containment access are administratively controlled. The airborne activity can be analyzed to determine processing (filtering) and release rates to continue to assure that 10 CFR 50, Appendix I, and 10 CFR 20 requirements are met.

- Instrument Line Break Inside Secondary Containment (UFSAR 15.6.2)

The reactor coolant instrument line break model described in the UFSAR Section 15.6.2 applies to the current evaluation of offsite dose impact. The off-site dose is a function of the mass release and the reactor coolant fission product source term. The normal reactor coolant fission product source term, including iodine activity, does not change due to power uprate. Since the reactor pressure does not increase and there has been no physical change to the instrument tubing, the mass and activity release rate does not change. Thus, power uprate does not change the dose values reported in UFSAR Section 15.6.2.

- Feedwater Line Break (UFSAR 15.6.6)

The 105% power uprate is conservatively expected to increase the total feedwater mass release and the amount of feedwater that flashes by 6%. The mass that flashes will increase by 6% from 165,000 pounds to 175,000 pounds. There is no fuel damage and the normal reactor coolant source term is not impacted by power uprate. The offsite doses could therefore increase by 6% based on the current model. The current values are less than 8.1E-09 rem whole body dose and 7.9E-07 rem thyroid dose. The small 6% offsite dose increase remains below one tenth of 10 CFR 100 limits.

- Radioactive Gas Waste System Leak or Failure (UFSAR 15.7.1)

The evaluation of this event is no longer a requirement of the current Standard Review Plan, NUREG-0800. The NUREG-75/087 radiological acceptance criterion for this event was 0.5 rem to the whole body or its equivalent to any organ from a GE determined design basis noble gas release of ≤100,000 μCi/sec. The analysis in UFSAR Section 15.7.1 states that there is no specific NRC guideline for this analysis.

A power uprate to 105% of the current licensed core thermal power has no impact on the radioactive gas waste system releases. The existing design basis concentrations of fission products in the reactor coolant are conservative and remain applicable. Therefore the radioactive gas source term is not dependent on reactor power level. The waste gas activity accumulation will continue to be administratively controlled as it is presently, following a power uprate to 3489 MWt.

A power uprate to 105% of the current licensed core thermal power will have no impact on the offsite doses due to a radioactive gas waste system leak or failure as described in UFSAR Section 15.7.1.

- <u>Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure (UFSAR 15.7.3)</u>

The radwaste equipment's radioactive inventory is a combination of the collection volume, the source stream(s) supply rates, and the activity in the source streams. For liquid radwaste following the 5% power uprate, all of these parameters will remain unchanged. The increase in the volume of processed condensate (which actually decreases the concentrations in the condensate) will increase the frequency of the resin processing, but it will not affect the physical operation (pumping rates, tank volume, and batch processing size). Equipment leakage, process sampling, decontamination operation, and housekeeping water requirements are not expected to change. Airborne radioactivity will continue to be processed by an HVAC system and mixed in the main stack before being released to the environment. Therefore, the radiological impact of a full waste concentrate tank rupture will not change.

The 5% power uprate will not have any impact on the postulated radioactive releases due to liquid radwaste tank failure described in UFSAR Section 15.7.3.

<u>Question 5</u>:

NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," indicates that for the fuel-handling accident, the release of I-131 could be 20% higher than those assumed in Regulatory Guide 1.25. Has the impact of increased gap fraction as a result of extended burnup fuel been included in the supporting analyses for the proposed change?

<u>Response 5</u>:

The impact of power uprate on the fuel handling accident consequences was evaluated by scaling the existing accident analysis in the UFSAR for the new source term expected for power uprate. The existing accident analysis was previously evaluated and determined to be bounding for the Siemens fuel assembly. The impact of extended burnup, including a larger (22%) I-131 gap fraction, was evaluated by Siemens and resulted in dose consequences that are a small fraction of the 10CFR100 limits. Since the extended burnup analysis was performed at a power level of 3910 MWt, which is higher than the uprated power, the conclusion that the dose consequences of a fuel handling accident are a small fraction of the 10CFR100 limits would also be applicable to the current power uprate.

<u>Question 6</u>:

Table A-1 of Attachment A to the July 14, 1999, submittal provides the loss-of-coolant accident (LOCA) radiological consequences. The second column is labeled "pre-uprate dose" and states that prior approved Amendment No. 125 incorporated the new source terms at uprated conditions. These dose numbers are not included in amendment number 125 to NPF-11. Please clarify the source of these dose numbers.

Response 6:

Superscript (a) in Table A1 is in error and should have referred to Amendment 126 of NPF-11 and Amendment 111 of NPF-18. Calculation L-001166 is included as Attachment C of Letter from F. Dacimo (ComEd) to U.S. NRC, "LaSalle County Station Units 1 and 2, Supplement to Application for Amendment of Facility Operating Licenses NPF-11 and NPF-18, Appendix A, Technical Specifications, Addition of a Ventilation Filter Testing Program," dated May 1, 1998.

The doses for calculation L-001166 are presented in Tables 13 and 14 of the calculation. These doses are for the extended burnup source term. See our response to Question 7 below.

The dose numbers for the Control Room and Auxiliary Electric Equipment Room (AEER) as presented in the Calculation L-001166 included only the whole body dose due to the immersion in the cloud of gamma emitters (0.1345 rem for the control room and 0.1242 rem for the AEER). These results were rounded to 0.2 rem for the NEDC-32701P, "Power Uprate Safety Analysis Report for LaSalle County Station Units 1 and 2," July 1999, (SAR) and license amendment. The total whole body dose also includes the direct radiation (or shine) from gamma radiation sources external to the facility. This second contribution was estimated to be 1.2 rem pre-uprate and 1.4 rem post-uprate for the submittal. The SAR and license amendment includes both contributions (see SAR Table 9-3). Calculation L-002523 was done after submittal of the SAR which calculated the direct dose as 1.24 rem.

Question 7:

What is the extended burnup source term power level used to calculate the LOCA radiological consequences?

Response 7:

The power level is not explicitly provided in Calculation L-001166. The extended source term data in L-001166 is from Letter No. JHR: 96:188, "Radioactive Release Analysis Source Term Values," from J. H. Riddle, Siemens Power Corporation, to R. J. Chin, ComEd, dated May 20, 1996. This is Reference 1 in Calculation L-001166.

The extended burnup source term in Calculation L-001166 is specified for 60,000 MWd/MTU, 2034 days of core residence, and 132.554 MTU/core. Therefore:

$$\frac{60{,}000\text{MWD}}{\text{MTU}} \quad \text{X} \quad \frac{132.554\text{ MTU}}{2034\text{ days}} \quad = \quad 3910\text{ MWt}$$