

**NORTHEAST UTILITIES**



NORWICH LIGHT AND POWER COMPANY  
HARTFORD ELECTRIC LIGHT AND POWER COMPANY  
MIDDLEBURY ELECTRIC COMPANY  
WATERBURY ELECTRIC COMPANY  
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December 15, 1978

Docket No. 50-336

Director of Nuclear Reactor Regulation  
Attn: Mr. R. Reid, Chief  
Operating Reactors Branch #4  
U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555

- References:
- (1) W. G. Council letter to R. Reid dated November 1, 1978.
  - (2) W. G. Council letter to R. Reid dated November 8, 1978.
  - (3) O. D. Parr letter to D. C. Switzer dated June 14, 1974; transmitting W. R. Stratton letter to D. L. Ray dated June 11, 1974.
  - (4) Supplement 1 to the SER dated March 7, 1975.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2  
Power Upgrading  
Proposed License Amendment

In Reference (1), Northeast Nuclear Energy Company (NNECO) provided a summary of the tasks involved in the application for a license amendment to operate Millstone Unit No. 2 at a higher power level, as well as a proposed schedule for submittal of the appropriate information to the NRC Staff. Subsequently, on November 21, 1978, a meeting between our respective Staffs was held to discuss the contents of Reference (1). This letter is being docketed in support of commitments made in Reference (1), and as a follow-up to certain items discussed during the meeting.

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company hereby proposes to amend its Operating License, DPR-65, by incorporating a change in Section 2.c(1) indicating that the steady-state reactor core power level shall not exceed 2700 megawatts thermal. This change is proposed to be effective commencing with the start of Cycle 3 operation, currently scheduled for May 15, 1979. The supporting changes of Appendix A of DPR-65 will be forwarded at a later date.

This application for an increase in the maximum allowable thermal output of the station is a fulfillment of the original design intent for Millstone Unit No. 2 as supported in the Final Safety Analysis Report. Section 1.1 of the FSAR states, in part, that:

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reactor thermal power of 2700 MWt. Attachment 1 merely updates the appropriate sections, including an update of the site parameters, and concludes that no new significant environmental impact would result. Also provided are an updated report on marine biology in the vicinity of the Millstone site, and an update of the environmental monitoring program description. These latter two updates are not directly related to the power uprating effort, but the reports are provided to more completely address relevant environmental topics.

Attachment 2, Evaluation of Radiological Consequences, addresses each of the following incidents:

- (1) Fuel Handling Accident in the Spent Fuel Pool
- (2) Waste Gas Decay Tank Rupture
- (3) Hydrogen Purge
- (4) Fuel Handling Accident in Containment
- (5) LOCA

For each of the above incidents, it has been concluded that for operation of Millstone Unit No. 2 at 2700 MWt, the radiological consequences are within the statutory limits of 10CFR100. The following incidents are not addressed in Attachment 2:

- (1) Steam Generator Tube Rupture
- (2) Main Steam Line Rupture
- (3) Control Rod Ejection
- (4) Seized Rotor
- (5) Control Room Dose to Operators

These latter five incidents cannot be addressed at this time as certain inputs to these calculations are not yet available from our NSSS vendor, Combustion Engineering. These incidents are currently being reanalyzed, and the results will be docketed concurrent with the submittal on non-LOCA safety analyses. This submittal remains scheduled for February 1, 1979.

Lastly, an Updating of Site Meteorology is provided as Attachment 3. This information serves as the basis for a key input to the radiological evaluations, and is provided such that NRC Staff review of this topic can be conducted in a timely fashion.

To further elaborate on the February 1, 1979 submittal of non-LOCA safety analyses, please note that the following transients/accidents will be addressed:

- (1) CEA Withdrawal Incident
- (2) Boron Dilution Incident
- (3) CEA Drop Incident
- (4) Loss of Flow Incident
- (5) Loss of Load Incident
- (6) Loss of Feedwater Flow Incident
- (7) Excess Load Incident
- (8) Steam Line Rupture Incident
- (9) CEA Ejection Incident
- (10) Steam Generator Tube Rupture Incident
- (11) Seized Rotor
- (12) LCS Depressurization

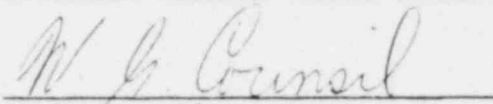
- (13) Loss of AC Power
- (14) Asymmetric Steam Generator Transients

Because all pertinent analyses and evaluations have not as yet been completed, a safety evaluation, pending successful completion of large and small break LOCA analyses, can be submitted with the February 1, 1979 letter.

We trust this information fulfills the commitments made in Reference (1) and during our November 21, 1978 meeting in Bethesda.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY



W. G. Council  
Vice President

Attachments

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