October 31, 2002

MEMORANDUM TO: James W. Andersen, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Travis L. Tate, Project Manager, Section 2 ***/RA/***
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: PILGRIM NUCLEAR POWER STATION - REQUEST FOR ADDITIONAL INFORMATION REGARDING APPENDIX K MEASUREMENT UNCERTAINTY RECOVERY - 1.5% POWER UPRATE REQUEST (TAC NO. MB5603)

Attached is a request for additional information transmitted by fax on September 20, 2002, to Mr. Bryan Ford of Entergy Nuclear Operations, Inc. (the licensee), regarding the Nuclear Regulatory Commission (NRC) staff's review of the licensee's application dated July 5, 2002, as supplemented August 29, 2002. The questions were transmitted to facilitate a discussion prior to the NRC staff's issuance of a formal request. Following receipt of the attached questions, the licensee informed the staff that they will provide the requested information. This memorandum documents the requested information transmitted to the licensee.

Docket No. 50-293

Attachment: As stated

CONTACT: Travis Tate, NRR/DLPM
301-415-8474

October 31, 2002

MEMORANDUM TO: James W. Andersen, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Travis L. Tate, Project Manager, Section 2 ***/RA/***
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: PILGRIM NUCLEAR POWER STATION - REQUEST FOR ADDITIONAL INFORMATION REGARDING APPENDIX K MEASUREMENT UNCERTAINTY RECOVERY - 1.5% POWER UPRATE REQUEST (TAC NO. MB5603)

Attached is a request for additional information transmitted by fax on September 20, 2002, to Mr. Bryan Ford of Entergy Nuclear Operations, Inc. (the licensee), regarding the Nuclear Regulatory Commission (NRC) staff's review of the licensee's application dated July 5, 2002, as supplemented August 29, 2002. The questions were transmitted to facilitate discussion prior to the NRC staff's issuance of a formal request. Following receipt of the attached questions, the licensee informed the staff that they will provide the requested information. This memorandum documents the requested information transmitted to the licensee.

Docket No. 50-293

Attachment: As stated

CONTACT: Travis Tate, NRR/DLPM
301-415-8474

DISTRIBUTION:
PUBLIC JAndersen TTate PDI-2 R/F LCox

ADAMS Accession Number: ML023040180

OFFICE	PDI-2/LA(A)	PDI-2/PM	PD1-2/SC(A)
NAME	LCox	TTate	JAndersen
DATE	10/25/02	10/29/02	10/29/02

OFFICIAL RECORD COPY

Request for Addition Information
Pilgrim Nuclear Power Station
License Amendment Request
Appendix K Measurement Uncertainty Recovery - 1.5% Power Uprate Request

1. Since the effects of flow-accelerated corrosion (FAC) on degradation of carbon steel components are plant specific, the values of the parameters affecting FAC; i.e. velocity and temperature changes, must be included. In addition, the corresponding changes in components wear rates due to FAC before and after the power uprate must be provided.

   A. Please provide the name and version of the predictive code used to project the need for maintenance/replacement of balance-of-plant (BOP) piping components prior to reaching minimum wall thickness requirements.

   B. Please provide the predicted change of wear rates calculated by the revised predictive code for the components most susceptible to flow-accelerated corrosion. Specifically, provide a detailed table with this information as illustrated below in a sample table:

System	**Description**	**% Change in Predicted Wear Rate**		**Change in Predicted Wear Rate, mils / year**	
		Average	Max	Average	Max
FW	Feedwater (FW) to FW Pump to High Pressure FW Heater	+0.46%	+0.19%	+0.008	+ 0.02

2. Attachment 2 to the amendment request, page S-1, second paragraph from the top, states,

   "This report follows the Nuclear Regulatory Commission (NRC)-approved format and content for Boiling Water Reactor (BWR) Thermal Power Optimization (TPO) licensing reports documented in NEDC-32938 P, "Generic Guidelines and Evaluations for General electric Boiling Water Reactor Thermal Power Optimization," called "TLTR."

   The above referenced document, NEDC-32938, has <u>not</u> yet been approved by the staff. Therefore, the above statement and all related references to NRC approval of this document should be removed from the submittal.

3. Please provide information relating to contingencies for an inoperable Crossflow UFM and effect of inoperable Crossflow UFM on thermal power measurement and plant operation. In this regard, also provide the following information:

   A. A proposed allowed outage time (AOT) for the feedwater flow instrument, along with the technical basis for the time selected.

   B. Proposed actions to reduce power level if the AOT is exceeded, including a discussion of the technical basis for the proposed reduction in the power level.

4. For all instruments that affect the power calorimetric, provide information to specifically address the following aspects of the calibration and maintenance procedures. The amendment request provides the required information only for the Crossflow UFM. Please provide it for the remaining instruments.

   c. Maintaining calibration
   d. Controlling software and hardware configuration
   e. Performing corrective actions
   f. Reporting deficiencies to the manufacturer
   g. Receiving and addressing manufacturer deficiency reports.

5. In reference to Section 2.5 of Attachment 2 to the amendment request, provide a summary describing the effect of the proposed power uprate on the structural integrity of the control rod drive mechanisms (CRDMs). Confirm that the existing design basis analysis for stress and fatigue cumulative usage of the CRDMs remains unchanged for the proposed 1.5 percent power uprate.

6. In Section 3.2.2 of Attachment 2 to the amendment request, you indicated that the effect of Thermal Power Optimization (TPO) was evaluated to ensure that the reactor vessel components continue to comply with the existing structural requirements of the ASME Boiler and Pressure Vessel Code. For the components under consideration, the 1965 Edition of the Code with addenda to and including Summer 1966, which is the construction code of record, was used as the governing Code. You also indicated that if a component underwent a design modification, the governing code for that component was the code used in the stress analysis of the modified component. Provide a summary of the components that were modified and the code editions/code cases (if applicable) other than the code of record that were used for the power uprate evaluation.

7. In Table 3-2 of Section 3.1, you indicated that the cumulative fatigue usage factor (CUF) for the feedwater nozzle is less than 0.8 for the current rated condition and less than 1.0 for the power uprate condition. Provide the actual calculated CUFs. Also, provide the allowable stress limits for reactor vessel components listed in Table 3-2. In reference to Section 3.3.2, provide a summary describing the effect of the proposed power uprate on the existing stress and fatigue analysis of the reactor internals. Also, provide comparison of calculated stresses and CUFs (similar to Table 3-2) for the limiting reactor internal components including allowable stress limits.

8. In reference to Section 3.5.3, you state that the original code of record for BOP piping and also for most safety related systems was ANSI B31.1. However, in your presentation on July 24, 2002, you indicated that design evaluation for SRV discharge line piping for the power uprate is in accordance with the requirements of ASME *B&PV Code* Section III 1977 Edition through Summer 1977 addenda. Provide the codes, code editions that were used for the RCPB piping and BOP piping for the power uprate.

9. In reference to Section 3.5.1, you indicated that the Response Spectrum Independent Support Motion (ISM) piping analysis methodology was applied for the SRVDL. At the July 24, 2002, meeting with the staff in Rockville, Md., you indicated the need to increase the SRV capacity as a result of the power uprate, and therefore, the current design margins of this piping system are reduced, due to the higher piping loads induced by the increased

SRV flow. The analysis for the SRVDL was therefore performed using the ADLPIPE computer code, the ISM methodology and Regulatory Guide (RG) 1.61 damping values. The ISM analysis used the square root of sum of squares (SRSS) approach for combining Group responses. This approach does not correspond to the staff position as stated in NUREG 1061, Volume 4. Likewise, the use of damping values in the analysis is not in accordance with the licensing basis for Pilgrim.

h. Provide the user manual, including the theoretical basis and bench-marking verification problems, for the ISM option in the ADLPIPE computer code.

i. Provide the justification for using the RG 1.61 damping values instead of the licensing basis damping values, as shown in the FSAR, for application with the ISM approach.

j. Provide the maximum stresses and CUFs for the SRVDL at critical locations, subjected to the following loading conditions:

(1) Current operating conditions.

(2) Power uprate operating conditions, using the uniform response spectrum piping analysis approach, and licensing basis damping values.

(3) Power uprate operating conditions, using the ISM piping analysis approach based on the Absolute Sum of Group responses, and licensing basis damping values.

(4) Power uprate operating conditions, using the ISM piping analysis approach based on the SRSS combination of Group responses, and licensing basis damping values.

(5) Power uprate operating conditions, using the ISM piping analysis approach based on Absolute Sum of Group responses, and RG 1.61 damping values.

c. Provide the licensing basis stress allowables.