Entergy

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November 21, 2002

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

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket 50-293
License No. DPR-35

Response to NRC Request for Additional Information
Appendix K Measurement Uncertainty Recovery – Power Uprate Request

REFERENCE: NRC Letter, "Request for Additional Information," dated November 13, 2002 (TAC No. MB5603)

LETTER NUMBER: 2.02.102

Dear Sir or Madam:

Entergy has reviewed the subject NRC request for additional information (RAI) dated November 13, 2002. Attachment 1 contains responses to the eight questions received.

Should you have any questions or comments concerning this submittal, please contact Bryan Ford at (508) 830-8403.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 21st th day of November 2002.

Sincerely,

Charles M. Dugger

JRH/dd

Attachment: 1. Response to NRC Request for Additional Information (9 pages)

A001

202102

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ATTACHMENT 1

LETTER NUMBER 2.02.102

Response to NRC Request for Additional Information

Appendix K Measurement Uncertainty Recovery-Power Uprate Request

1. NRC Request: In Section 10.5 "Operator Training and Human Factors" of the Pilgrim Nuclear Power Station (PNPS) Thermal Power Optimization (TPO) uprate amendment, the licensee has not identified any changes to the control room (i.e., controls, displays, and alarms) that will be available to the operators. The U.S. Nuclear Regulatory Commission (NRC) staff anticipates that modifications will be made to the control room and the simulator to assist operators in identifying problems with the ultrasonic flow measurement instrumentation installed to support the TPO uprate. The licensee is requested to provide a description of any modifications, their use and controls that will be in place to properly monitor them. Additionally, the licensee is requested to provide a brief description of any effects of the power uprate on the safety parameter display system. Finally, please provide a commitment to identify and revise all plant operating procedures that will be affected by the power uprate prior to operation above the current licensed thermal power level. Please use "NRC Regulatory Issue Summary 2002-03: Section VII, Questions 2, 3, and 4 to frame your response and provide the appropriate information.

Response

The PNPS power uprate submittal (Reference letter 2), provided a discussion of operating and maintenance procedures to be developed or revised to support implementation of TPO. Attachment 2 of this letter also included several PNPS commitments regarding procedure development and revision and completion of all activities necessary to support TPO prior to exceeding current licensed thermal power. As stated in regulatory commitment C2, all changes necessary to support operation above the current licensed thermal power will be completed prior to exceeding 1998 MWt.

Additional information regarding operator training and human factors is provided below.

Changes will be made to the control room and training simulator alarms and displays to alert operators of off-normal conditions associated with the proposed power uprate; and to assist them in resolving any problems with the new equipment. New procedures will be developed and where appropriate, existing ones revised. The plant operators will be trained in the use of the new system before implementation of Power Uprate, and all required modifications will be completed prior to implementation of Power Uprate.

Pilgrim has reviewed its systems, and conducted a thorough review of industry experience and believes it has identified all necessary modifications associated with the power uprate. Changes in procedures (normal, emergency and abnormal), and in operator training, both classroom and simulator, are being made to ensure that changes in operator actions do not adversely affect the plant's commitment to defense in depth or its safety margins.

The control room and the simulator alarms and displays will be modified as follows:

1) A new annunciator window will be installed and engraved with FW FLOW CORRECTION FACTOR TROUBLE ALARM.

2) New Emergency and Plant Information Computer (EPIC) Displays will be developed to show the status of the Correction Factor (Cf) and the AMAG installation. The display will include switches to initiate or remove the correction factor and to switch between the two independent trains of AMAG.

<u>Annunciator Window Trouble Alarms</u>
The EPIC Computer will be modified to provide control room annunciation of FW Flow Correction Factor Trouble conditions. This alarm will be designed to actuate when any of the following "Trouble" conditions exist:

1) **If EPIC loses valid communications or data from AMAG** – This condition may be caused by any number of internal AMAG faults or conditions which would indicate that either or both AMAG "A" and "B" are not functioning correctly and that troubleshooting may be required.

2) **If the Correction Factor is bad quality** – This condition is due to not satisfying all criteria during the calculation of the correction factor.

3) **Venturi rapid defouling** – This condition is due to a rapid change in the venturi calibration due to a process transient which causes any build-up on the venturi surface to release.

4) **Turbine first-stage Pressure / Steam flow / Feedwater Flow Mismatch** – This condition is due to a change in the ratio of turbine first-stage pressure/Feedwater flow and Feedwater/Feedwater flow. This condition may be indicative of a rapid change in the venturi calibration.

5) **EPIC Correction Factor timed–out** – This condition exists when the time period allowed for operation with the Correction Factor applied without verification has expired. The Correction Factor must be removed.

6) **Feedwater flow change during EPIC generated correction factor** – This condition exists when there is a feedwater flow change during the timed period allowed for operation with the Correction Factor applied without verification.

EPIC Screens
The annunciator will alert the operator to problems in the system. New EPIC screens will provide details of the problem and include "switches" to permit the operator to take appropriate action.

The new EPIC screens will have two "Operator controlled software switches," one for initiation/ removal of the CROSSFLOW Correction Factor and the other for switching between AMAG A and B systems. The CROSSFLOW Correction Factor will only be implemented or removed by an Operator Controlled Software Switch. Note: EPIC is presently programmed for manual determination and implementation of the Correction Factor.

EPIC Plant Computer displays will include CROSSFLOW System related parameters, including as a minimum:
- CROSSFLOW System status (available, in use, not in use, quality etc.).
- The correction factor.
- Quality point status.
- Manual input data.
- Correction factor status in the feedwater flow input adjustment (e.g., implemented/not implemented).

- Parameters such as the EPIC points that compare first-stage pressure and steam flow with feedwater flow are to be displayed on an Emergency and Plant Information Computer display.

An EPIC Plant Computer display shall be available to aid in CROSSFLOW System correction factor and CROSSFLOW System data link troubleshooting and maintenance.

The Crossflow Cf is updated from the average of readings in a long buffer. This is typically 4 hours worth of data. However, depending on the quality of the data, the collection time could be 12 hours.

An alarm is initiated if the data in the short buffer deviated by a set amount from the long buffer. The short buffer is expected to contain ½ to one hour's worth of data.

Flow data is updated with a latency of no more than 60 seconds.

Safety Parameter Display System

Three screens on the SPDS will be changed to reflect the new blowdown rates from the enlarged SRVs.

- Displays 37 and 38 are the **Heat Capacity Temperature Limits**
- Display 39 is the **Boron Injection Initiation Temperature**

2. NRC Request: In section 7.1 "Turbine Generator" of the PNPS TPO uprate amendment, PNPS stated that the overspeed trip settings will be changed from 110% and 111% to 110.6% and 111.6%. PNPS specified increased entrapped energy within the turbine as the basis for the changes. Please provide a description of the effects of these changes on any transients or anticipated operational occurrences that credit the overspeed trip in the safety analysis. The licensee is requested to provide information to demonstrate that the plant will continue to operate within its design limits.

   Response

   Since the PNPS TPO uprate amendment submittal, the turbine overspeed trip settings and basis for these settings have been further evaluated. The GE Power Systems group has performed additional analysis for the overspeed sensitivity and recommended that the actual overspeed trip settings remain unchanged.

   Based on this recommendation, Pilgrim has elected to not change the turbine overspeed trip setpoints.

   The uprate amendment submittal as initially written is incorrect in that the overspeed trip settings stated in the NRC letter dated November 13, 2002, are in fact the allowable range for the mechanical overspeed setpoint. It is desirable to have the setpoint range above the calculated value of normal overspeed (NOS) but is not a requirement. The range increases slightly with the increase in entrapped energy with the actual setpoint remaining unchanged.

   The turbine overspeed trip devices (emergency governor mechanical overspeed and backup overspeed (BOS) trip) are not credited in the transients or anticipated operational occurrences that have been evaluated for Pilgrim. The two transients that are evaluated are Generator Load Reject without Bypass and Turbine Trip without Bypass. These transients credit the turbine trip (No. 1 vacuum trip, closes control and stop valves) and the acceleration relay (closes control valves) with limiting turbine overspeed as each will initiate sooner and have a faster response time. The turbine overspeed trip devices are not expected to actuate during these transients even with the small increase in entrapped energy. The time for valve closure is not affected by the increase in entrapped energy.

The following paragraphs provide additional technical information.

The mechanical overspeed (MOS) trip setpoint is derived based a specific turbine's unique potential to accelerate under no load due to transient and entrained energy downstream of the closing valves.

Overspeed calculations are made to insure that the MOS trip setting is set appropriately to insure that if a full load rejection occurs and a turbine trip (No. 1 vacuum trip) is not actuated from the generator lock-out relay, the MOS trip is not activated. This calculation is referred to as the normal overspeed (NOS). In addition, the calculated overspeed from the highest MOS trip setting must remain below 120%, as this is the upper limit of the system design. The backup overspeed (BOS) trip provides a backup to the MOS trip.

During a full load rejection, the unit can speed up due to the following two factors:

1. The time delay for the valves to close and control system lags.

2. The entrapped steam in all of the attached piping can still flow through the unit and cause the speed to increase.

GE records indicate the original turbine was designed and installed with the following overspeed data:

Normal Overspeed (NOS) = 109
Recommended MOS trip setting range = 110% -111%
Backup Overspeed (BOS) Trip = 112% +/-1%

For the uprate steam conditions the calculated overspeed conditions are as follows:

Normal Overspeed (NOS) = 109.8%
Maximum Mechanical Overspeed (MOS) Trip Setting = 111.8%
Backup Overspeed (BOS) Trip = 112% +/-1%

The current MOS trip settings are below the maximum and therefore do not require change.

The NOS has been calculated to be slightly higher than the original, 109.8% as compared to 109%, and is therefore closer to the current OS trip setpoint. This means that the margin between the calculated overspeed and the overspeed trip setpoint for the uprate condition is smaller. This is acceptable. A load reject (expected event) also trips the stop and intermediate stop valves through the generator lockout relay via the No. 1 vacuum trip. A turbine overspeed trip is not expected from a load reject event.

FSAR section 7.11.3.3.4 has an NOS calculated value of approximately 108%. This will be changed to 109.8% (Reference: FSAR sections 7.11.3.3.3 and 7.11.3.3.4)

3. NRC Request: In Section 7.3 "Turbine Steam Bypass" of the PNPS TPO uprate amendment, the licensee stated that the steam-flow capacity of the bypass system was 25% of the 100% rated flow at current licensed thermal power. The licensee specified that the percentage of bypass at TPO uprate conditions would decrease even though the capacity would remain unchanged. Additionally, the licensee specified that some transient analyses credit only the actual capacity. Due to higher rated thermal power, this may result in unintended consequences (i.e., higher peak reactor vessel pressures and temperatures) for transients crediting the operation of this system. The licensee is requested to verify that a review of all transient analyses that credit the actual capacity of this system has been performed and identify whether it is sufficient to prevent exceeding safety and design limits.

   Response

   Section 7.3 of the Pilgrim TPO Safety Analysis Report (TSAR), NEDC-33050P, states that the transient analyses that credit the turbine bypass system availability use the actual capacity. Further, Section 7.3 states that the transient analysis results presented in Section 9.1 of the TSAR are acceptable.

   The transient analysis results presented in Section 9.1 are based on the generic evaluations of Appendix E of the TPO Licensing Topical Report (TLTR), NEDC-332938P. The evaluations and conclusions of Appendix E are applicable to PNPS. The availability of the turbine bypass was credited in transient event, FW Control Fails - Max Dmd (Demand), presented in Table E-2, consistent with the standard approved GESTAR methods.

   The process for evaluation of the transient analysis does not verify the safety and design limits at the time of the plant-specific license amendment. Rather, the sensitivity studies described in Appendix E of the TPO Licensing Topical Report (TLTR), NEDC-332938P, has shown that the effect of TPO is small. The affect of TPO on the limiting events will be confirmed as part of the normal reload analysis.

4. NRC Request: In Section 6.4.5 "Ultimate Heat Sink" (UHS) of the PNPS TPO uprate amendment, the licensee stated that "... TPO operation increases the amount of heat discharged to the UHS by a small amount..." The staff reviewed this statement and the original 1972 Final Environmental Impact Statement for PNPS. The staff requests the licensee provide either of the following: 1) A statement identifying the environmental assessment performed for PNPS, and approved by the NRC, that demonstrates the heat load rejected to the UHS under the TPO uprate conditions is bounded, or 2) A technical justification for the increased heat load rejected to the UHS including relevant numerical data such as a change in heat load discharged, effluent temperature, and condenser cooling water flow rate.

   Response

   Pilgrim Nuclear Power Station's thermal discharge to the ultimate heat sink is regulated by the facility's National Pollutant Discharge Elimination System (NPDES) Permit. This permit limits the maximum discharge temperature to 102°F and limits the differential temperature (ΔT_c) across the main condenser to 32°F. The permit is currently in the process of being renewed and it is anticipated that the limits will remain the same.

Pilgrim has performed calculations based on current and new General Electric heat balances which indicate that Thermal Power Optimization (TPO) will result in an approximate 0.8°F maximum increase in ΔT_c. An increase of this small magnitude will not affect Pilgrim's ability to operate within the current and anticipated NPDES permit requirements for thermal discharge to the ultimate heat sink.

5. NRC Request: The American Society of Mechanical Engineers overpressure analysis was performed at 102% power, which would bound the proposed uprate power level. What was the calculated peak pressure for the ASME overpressure analysis for the current cycle, based on the current safety relief valve (SRV) capacity?

   Response

   The calculated peak reactor pressure for the current cycle, 102% reactor thermal power and existing SRV capacity is 1301 psig. This is for the MSIV Closure (Flux Scram) Event (Ref. J11-03808-10SRLR, Rev. 0, Supplemental Reload Licensing Report for PNPS, Reload 13 Cycle 14).

6. NRC Request: Section 1.2.1, "TPO Analysis Basis," states that the throat size of the SRVs is being increased which may result in a 7.5% increase in the SRV capacity. Table 9-1, "Key Inputs for ATWS [Anticipated Transient Without Scram] Analysis," indicates that the peak ATWS pressure is 1495 psig, based on the TPO conditions and the increased SRV capacity. The analysis used SRV setpoint drift of 22 psig. What type of SRVs are installed at PNPS? Explain the basis of the SRV drift value used in the analysis.

   Response

   The Pilgrim SRVs are Target Rock, 2 Stage, Model 7567F, 6" x 10" valves. Setpoint tolerance, per the Technical Specifications, is ± 1% (± 11 psi) based on the setpoint of 1115 psig. The ATWS analysis assumed Technical Specification upper limit plus one SRV was set at 1136 psig (see Table 1). A drift uncertainty, about the mean, of 22 psid (1δ) was assumed.

Valve		Valve Setpoints (psig)		Analysis Basis Setpoints (psig)	
Number	Group	Nominal Trip Setpoint	Tech. Spec. Upper Limit	Mean	With Drift Uncertainty
1	1	1115	1126	1136	1136
2	2	1115	1126	1126	1144.6
3	2	1115	1126	1126	1126
4	2	1115	1126	1126	1107.4
Group Average Setpoint (psig)				1128.5	1128.5

Table 1: SRV Setpoints

The average as-found setpoint deviation from 1987 though 2001 is + 0.92% (10 psi). This results in an average as-found setpoint of 1125 psig. During RFO 12 (1999), two SRVs had large errors in the setpoint (+9.15% and 5.83%) due to pilot sticking that was caused by a crud burst in the reactor vessel related to a chemical decontamination and subsequent flood up of the relief lines before the pilot assemblies were removed. These two data points are discounted in the calculation of the average setpoint deviation. Based on Table 1, the average setpoint used in the ATWS analysis is 1128.5 psig which bounds the average as-found setpoint of 1125 psig. Therefore, the calculated ATWS transient pressure of 1495 psig is conservative.

7. NRC Request: Section 4.3 of the submittal states that the pre-TPO SAFER/GESTR-LOCA analysis did not have sufficient margin to the statistical Upper Bound peak clad temperature (PCT) limit of 1600°F, a plant specific analysis for PNPS was performed at the TPO rated thermal power level. What is the calculated Upper Bound PCT for the TPO conditions?

   Response

   The TLTR requires at least 10 degrees of margin to the Upper Bound PCT or re-evaluation of the Upper Bound PCT. The calculated Upper Bound PCT for the current licensed thermal power is less than 1580°F for GE14 fuel and 1599°F for GE11 fuel. The increase in Upper Bound PCT at TPO RTP is less than 1°F. Therefore, at the TPO RTP level the Upper Bound PCT remains below 1600°F for both fuel types.

   In order to satisfy the original requirements of the SAFER/GESTR-LOCA analysis method to maintain the Upper Bound PCT ≤ 1600°F, the permissible MAPLHGR for both GE14 and GE11 fuel types is restricted to a level below the design power limit of the fuel. To meet the Upper Bound PCT limit, the MAPLHGR is limited for both fuel types at current licensed thermal power (CLTP) and at TPO Rated Thermal Power.

   It is noteworthy that the 1600°F Upper Bound PCT limit is specific to the SAFER/GESTR-LOCA analysis methodology developed by General Electric. Since the TLTR was prepared, the SAFER/GESTR-LOCA methodology has been revised by removal of the requirement to perform plant specific calculations that demonstrate compliance with the Upper Bound PCT limit of 1600°F [Reference 1]. With the removal of the plant specific Upper Bound PCT limit, the SAFER/GESTR-LOCA methodology acceptance criteria is based solely on 10 CFR 50.46 requirements. In the next operating cycle, MAPLHGR limits will be established to ensure 10 CFR 50.46 requirements are met without consideration for the previous Upper Bound PCT limit.

   References: GESTR-LOCA and SAFER Models for Evaluation of Loss-of Coolant Accident, Volume III, Supplement 1 – Additional Information for Upper Bound PCT Calculation, NEDE-23785P-A, Supplement 1, Revision 1, March 2002.

8. NRC Request: Section 3.9, "Reactor Core Isolation Cooling," evaluates the reactor core isolation cooling (RCIC) system capability. The submittal states that the generic evaluation in the Thermal Power Optimization Licensing Topical, NEDC – 32938P Section 5.6.7 is applicable to PNPS. Confirm that the loss of all feedwater event was performed at 102% power. (This event is the design basis assumption for the RCIC system.)

Response

The original Loss of Feedwater (LOFW) analysis was performed at 100% power. The generic evaluation provided in NEDC-32938P, Licensing Topical Report Thermal Power Optimization (TLTR), is not applicable to Pilgrim Nuclear Power Station, contrary to the statement made in Section 3.9 of the Pilgrim TPO Safety Analysis Report (TSAR). However, the TPO uprate does not affect the RCIC system operation, initiation, or capability requirements.

To support the PNPS TPO application, an evaluation was performed to assess the effect of a 2% increase in power for the LOFW analysis. The evaluation made use of previous generic analyses for the Loss of Feedwater performed for BWR3s to determine the range of core power over which the RCIC system can fulfill its design objective. The results of this generic evaluation were used to estimate the effect for Pilgrim with a 2% increase in power. The safety criterion for this event is that the RCIC (the smaller of the two high pressure coolant supply systems) shall maintain sufficient water level inside the core shroud to assure that the top of the active fuel remains covered throughout the event. The results of the evaluation indicate that the effect of a 2% power increase is small (~4 inch decrease in the minimum level), and there is still at least 4 feet of margin above the Top of Active Fuel (TAF) for Pilgrim. Therefore, the safety criterion is met, and the Loss of Feedwater flow event is acceptable for the 2% increase in core power. This evaluation bounds the proposed 1.5% increase in core power for the Pilgrim TPO uprate.

REFERENCES:

1. NRC Fax, "Request for Additional Information," dated September 20, 2002
2. Entergy Letter 2.02.048, dated July 5, 2002, Appendix K Measurement Uncertainty Recovery – Power Uprate Request
3. Entergy Letter 2.02.080, dated August 29, 2002, Appendix K Measurement Uncertainty Recovery – Power Uprate Request Submittal of Non-Proprietary Version of TSAR
4. Entergy Letter 2.02.087, dated September 27, 2002, "Response to NRC Request for Additional Information, Appendix K Measurement Uncertainty Recovery – Power Uprate Request
5. Entergy Letter 2.02.096, dated November 6, 2002, "Response to NRC Request for Additional Information, Appendix K Measurement Uncertainty Recovery – Power Uprate Request