FENOC

FirstEnergy Nuclear Operating Company

Perry Nuclear Power Station
10 Center Road
Perry, Ohio 44081

Mark B. Bezilla
Vice President

440-280-5382
Fax: 440-280-8029

September 17, 2008
L-08-270

10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Response to Request for Additional Information Regarding Relief Request IR-054, Revision 0 (TAC No. MD8458)

By a letter dated July 31, 2008, the Nuclear Regulatory Commission (NRC) staff requested additional information related to Relief Request IR-054, Revision 0, which is a request for the Perry Nuclear Power Plant for relief from certain Inservice Inspection requirements associated with the implementation of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The response to the staff's Request for Additional Information (RAI) is attached.

On September 12, 2008, the NRC staff requested further information as part of the RAI response and it was agreed that a submittal date beyond 45 days for the response would be acceptable.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at (330) 761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 17, 2008.

Sincerely,

Mark B Bezilla

A047
NRR

Attachment:
Response to Request for Additional Information Related to Relief Request IR-054, Revision 0

cc: NRC Region III Administrator
NRC Resident Inspector
NRR Project Manager
Utility Radiological Safety Board

The following supplemental information is provided to respond to a Request for Additional Information (RAI) that was provided on July 31, 2008. The NRC question is repeated below, in bold, and is followed by the FirstEnergy Nuclear Operating Company (FENOC) response for the Perry Nuclear Power Plant (PNPP).

Background

The technical bases supporting your request for relief from the ASME Code, Section XI examination requirements regarding RPV nozzle-to-vessel welds and nozzle inner radii and use of an alternative based on ASME Code Case N-702, "Alternative Requirements for Boiling- Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds," for Perry are documented in BWR Vessel and Internals Project (BWRVIP) Report BWRVIP-108, "Technical Basis for the Reduction of Inspection Requirements for the Boiling-Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii." The safety evaluation (SE) on BWRVIP-108 dated December 19, 2007, listed conditions for an applicant to demonstrate the plant-specific applicability of BWRVIP-108 to its plant. Your submittal indicated that all conditions specified in the SE are satisfied. One of the conditions is that the RPV heatup/cooldown rate is less than 115° F. However, the information from the NRC resident inspector at Perry indicated that, recently, Perry had more than two events with the heatup/cooldown rate exceeding 115° F.

Request For Additional Information

Please discuss heatup/cooldown rate versus time information for these events and their frequency (how often does it occur). The staff plans to use this information to adjust the probabilistic fracture mechanics results reported in BWRVIP-108 to assess its impact.

Response to Request:

The table provided on the following pages lists those PNPP transient events that have occurred over the past ten years that have exceeded a heatup/cooldown rate of 115° F. This table lists the date of each event, the temperature measurement location, the maximum heatup/cooldown temperature rate, and corresponding comments, which provide additional clarifying information about each event.

FENOC believes that the four transient events detailed in the following table do not invalidate the plant-specific applicability of the BWRVIP-108 report for the PNPP nor does this data conflict with the technical basis to incorporate Code Case N-702 provided in Relief Request IR-054, Revision 0.

Key: **CD = Cool Down**
HU = Heat Up
BH = Bottom Head
BHDN = Bottom Head Drain

Date	Event	Measurement Location	Maximum rate of change in temperature	Comment
04/29/2001	CD	Reactor Recirculation Pipe Loop A	-220 (°F/hr) +190 (°F/hr)	Manual scram. Reactor Recirculation pumps tripped. Severe reactor recirculation pipe transients. Reactor pressure vessel temperature rates measured from bulk saturation temperature remained within the 100°F/hr limit. Plant computer data used to determine maximum rates for reactor recirculation piping.
07/12/2001	CD	BHDN	-200 (°F/hr)	Plant scram as a result of a loss of Feedwater. BHDN cool down. No other components were noted as affected. Reactor recirculation temperature did not oscillate. Reactor pressure vessel temperature rates measured from bulk saturation temperature remained within the 100°F/hr limit.

Date	Event	Measurement Location	Maximum rate of change in temperature	Comment
12/15/2001	HU/CD	BHDN	-257 (°F/hr)	Plant scram as a result of a loss of Feedwater. BH, BHDN, and reactor recirculation pipe experienced temperature changes in excess of 100 °F/hr. Plant computer data used to determine maximum rates for reactor recirculation piping. Reactor pressure vessel temperature rates measured from bulk saturation temperature remained within the 100°F/hr limit.
		BH	-115 (°F/hr)	
		Reactor Recirculation Pipe Loop B	+190 (°F/hr)	
			-177 (°F/hr)	
11/28/2007	CD/HU	BHDN	+259 (°F/hr)	Plant scram as a result of a loss of Feedwater. Post scram, loss of Reactor Water Clean Up system and no vessel recirculation flow - just natural circulation. BHDN rate from surveillance data for one hour. Recirculation temperatures from plant computer data. Reactor pressure vessel temperature rates measured from bulk saturation temperature remained within the 100°F/hr limit. Additional discussion of this event is provided on the following page.
		Reactor Recirculation pipe Loop A	-268 (°F/hr)	
		Reactor Recirculation pipe Loop B	+263 (°F/hr)	

For the November 28, 2007 plant scram, the following was concluded after review and evaluation of plant operating data from this transient event:

1. The heatup/cooldown rate for the vessel head, bottom and shell flange were well within the 100 °F/hr limit of PNPP Technical Specification (TS) 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

2. The maximum heatup/cooldown rate for the bottom head drain and recirculation loop A/B nozzle exceeded 100 °F/hr. However, a report by a technical consultant, Stevenson & Associates, provided an evaluation demonstrating that the impact of heatup/cooldown rates in excess of those stated in PNPP TS 3.4.11 is acceptable and the usage factor for the affected reactor pressure vessel region and components is less than 1.00.

Therefore, for the November 28, 2007 plant scram, the reactor pressure vessel and all the affected components were determined to continue to meet the requirements of American Society of Mechanical Engineers (ASME) Code and are capable of performing their design function.