September 27, 2000

Mr. Scott Portzline
3715 N 3rd Street
Harrisburg, PA 17110

SUBJECT: YOUR CONCERNS WITH THE REVISED REACTOR OVERSIGHT PROGRAM

Dear Mr. Portzline:

This letter responds to the concerns expressed in your letter dated August 11, 2000, in which you had follow-up questions related to the public meeting held in the vicinity of Three Mile Island (TMI) on August 2, 2000. The purpose of that meeting was to explain our new reactor oversight process and how it is being used at nuclear power plants. In preparing for the meeting I ensured that both an NRC headquarter's representative and an NRC Region I senior manager participated, in order to gain feedback related to any concerns which may be raised.

On August 24, 2000, I called you to confirm that I had received your letter. During our phone conversation you reiterated your concerns regarding the lack of aggressiveness of NRC to establish new requirements that define specifically when a licensee is unsafe and must be ordered to shut down, whether the NRC has sufficient resources to find safety issues, and your belief that the public had little chance to shape the new regulatory process.

I appreciate the time and effort you made to attend our meeting and document your thoughts and insights. Your letter was provided to the appropriate NRC staff responsible for the program area development for their consideration. The responses below are intended to address your questions and provide additional information. Overall, I do believe that the new regulatory process is an enhancement to our methods and provides a sound basis for the NRC's oversight efforts.

On the subject of public participation shaping the new regulatory process, the revised reactor oversight process (ROP) has been under development since directed by the Commission in SECY-98-045, Staff Requirements Memorandum, dated June 30, 1998. The NRC solicited members of the public, industry representatives, and representatives of public interest groups, to provide comments during the past two and one half years regarding design, development, pilot testing, and implementation of the new reactor oversight process. During this period the NRC gained additional public input during a public comment period and at various public meetings and workshops held in each region. A meeting to introduce the revised ROP was conducted in the vicinity of each pilot reactor site at the beginning and end of the six month pilot period to both inform and gather public feedback. During the pilot period, several other meetings were conducted with stakeholders to review specific aspects of the new process and in January 2000 a lessons learned public workshop was conducted. Information about the new process has been posted on the NRC web site (www.nrc.gov/NRR/OVERSIGHT/index.html) for the purpose of keeping those interested public informed so they can provide meaningful input. As part of the initial implementation, meetings, such as the one you attended on August 2, are being held at each of the 65 sites with local officials and the public to formally introduce the new program and allow direct questioning. Further, we are currently planning to hold an all-day meeting on December 13, 2000, in the King of Prussia area, at which time representatives from industry and NRC will have an opportunity to share their experiences thus far with the new process. Our current view is for the meeting to promote discussion between a panel and those in attendance. We will issue a meeting notice when the plans are finalized and the meeting will be open to the public. We will send you a copy of the notice. In addition, specific comments regarding any facet of the new reactor oversight program can be submitted to Mr. William Dean, NRC Inspection Program Branch - Mail Stop 7A15, Washington, D.C. 20555-0001.

Your concern that the new process will not clearly define when a plant is unsafe and must be shut down for repairs is addressed in two ways. As you know, each plant's license incorporates Technical Specifications which include detailed requirements for equipment and includes actions, such as plant shutdown, that are required to be taken when equipment is not capable of performing its intended safety function. Secondly, NRC actions based on a licensee's overall safety performance are addressed through the NRC Manual Chapter (MC) 0305, Operating Reactor Assessment Program. This Manual Chapter describes the Action Matrix which was developed to provide guidance for consistent consideration of actions. Action decisions are triggered directly from the threshold assessments of performance indicators (PIs) and cornerstone inspection areas. This provides pre-established criteria for agency decisions, thus providing for consistent and predictable NRC response to licensee performance. It remains true that NRC senior management deliberations would be needed in order for the Agency to ultimately decide to cause shutdown of a plant that was still meeting its license requirements; however, our ROP provides an improved set of assessment tools for evaluating the effect on safety margins of licensee performance weaknesses.

Regarding emergency exercises, inspection findings from the NRC biennial exercise evaluations are assessed using the significance determination process for the emergency preparedness area. The emergency response cornerstone would also be assessed in accordance with the MC 0305 process. The regulatory tool the NRC would use to determine if a plant shutdown or other enforcement action is appropriate for emergency preparedness deficiencies is 10 CFR 50.54(s)(2)(ii). It states that, "if the NRC finds that the state of emergency preparedness does not provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency and if the deficiencies are not corrected within four months of that finding, the Commission will determine whether the reactor shall be shut down..." Significant emergency preparedness exercise deficiencies would lead to a programmatic review, which could result in an ordered plant shutdown if reasonable assurance could not be provided.

You are correct that licensees are not legally required to report PI data. All licensees volunteered to report PI data in order for the new oversight process to be implemented at their plants. The use by industry of voluntary initiatives in the regulatory process was approved by the Commission in SECY-99-063. In doing so, the Commission specified that voluntary industry initiatives will not be used in lieu of regulatory action where a question of adequate protection of public health and safety exist. Once PI's are reported, the NRC requirements regarding accuracy of reported information are applicable and enforceable. Should a licensee decide not to submit PI information, the NRC has in place an inspection procedure and would use its inspection resources to gather the information in order to complete the assessment program defined in MC 0305. In addition, the baseline inspection program calls for NRC inspectors to verify the submitted data for accuracy and consistency for each PI.

As part of the initial implementation period, the question of having sufficient resources to conduct the new inspection program has been under review. The baseline inspection program is risk-informed and identifies the minimum level of inspection at a plant (regardless of performance) in order for the NRC to have sufficient information to determine whether plant performance is at an acceptable level. Your request for a comparison of inspection hours per quarter prior to the new system and afterwards for all Pennsylvania plants cannot be done until sufficient time has elapsed to conduct the new program. Further information regarding NRC resources to conduct the program will be reported to the Commission following the first year of initial implementation.

Similarly, it is too soon to provide a comparison of safety issues found before and after implementation of the new system. However, as part of the development of the new program, feasibility reviews of the inspection finding risk characterization and reactor oversight processes were performed using known safety issues from previous inspection activities. These reviews concluded that the new process was feasible to pilot because the agency actions that had been taken in response to plant performance were similar to those expected under the new program. Additionally, during the 6-month pilot program, another feasibility review of non-pilot plant events and follow-up inspections was done and concluded that the new processes were capable of adequately determining agency response to events and plant conditions. These reviews are discussed in SECY-99-007A and SECY-00-49, which are available on the NRC web site. Overall, the new oversight process should better address your concern regarding plants operating with a known safety issue, by being more risk informed, which will draw more attention to those issues of more risk significance. In the new program, each identified safety issue is evaluated by the NRC to determine its significance (both safety and risk). For these identified safety issues, the NRC actions should be more consistent and predictable.

While you expressed concerns that the inspection process would not have prevented the TMI-2 accident, it is important to recognize that we have revised our oversight process to be more risk informed such that the inspection staff can devote its attention to the more significant issues at each plant. As you know, after the TMI-2 accident, licensees were required to implement numerous changes (administrative and physical) to address many issues that resulted from post-accident reviews. Our new inspection process continues to look, in part, at the most risk significant of these changes, and others, to ensure that licensees are meeting the regulatory requirements. The new assessment process is designed to guide agency actions in a more consistent manner.

I appreciate your interest in our program and look forward to your input and participation in the developmental efforts of the NRC reactor oversight process. Should you have any additional questions, or if the NRC can be of further assistance, please call me at 610-337-5146.

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch 7
Division of Reactor Projects

Enclosure: NRC Manual Chapter 0305, Operating Reactor Assessment Program

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