From: "PMBLANCH@aol.com" ("PMBLANCH@AOL.COM") PM Blanch public citizen
To: JAZWOL@aol.com, JA Zwolinski NRC
Date: Saturday, September 30, 1995 4:19 pm
Subject: 9/28/95 (SMTP Id#: 55974)

Mail included in attachment.

CC: dan_berkovitz@epw.senate.gov,

9510050153 951002
PDR ADOCK 05000287
H PDR

JOHN:
This will confirm my total displeasure with your response. (copy enclosed).
I will be responding formally to the question related to operability for the condensate pots.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 28, 1995

Mr. Paul M. Blanch
135 Hyde Road
West Hartford, CT 06117

Dear Mr. Blanch:

This letter is in response to your INTERNET electronic mail (e-mail) of March 16, 1995, in which you asked if the New York Power Authority (NYPA) and the NRC had verified that every Indian Point 3 motor-operated valve (MOV) was operable as required by the plant's Technical Specifications (TSs), and to your letter of March 29, 1995, in which you responded to my letter of March 9, 1995. In the March 29, 1995, letter you requested that the NRC investigate whether every boiling-water reactor (BWR) licensee has performed operability determinations of its reactor vessel level condensing pots and whether every pressurized-water reactor (PWR) licensee has performed operability determinations of its pressurizer level condensing pots. You asserted that each licensee is required to perform such determinations in accordance with 10 CFR Part 50, Appendix B, Criterion XVI, and Generic Letter (GL) 91-18. You also asserted that the NRC refuses to investigate serious safety violations and that my letter of March 9, 1995, was incorrect regarding an Inspector General (IG) investigation.

In answer to your first concern, NYPA has indicated that MOVs subject to the NRC GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," program are operable. In addition, the NRC has conducted an inspection in this area, and found that NYPA has satisfactorily completed a review of safety-related MOVS.

With respect to the Indian Point 3 MOV program, Enclosure I to this letter contains a brief overview of NYPA's GL 89-10 program and a status update of the Indian Point 3 MOV program. As part of the Indian Point 3 restart inspection effort, the NRC staff reviewed the MOV program. Some weaknesses were detected in the MOV program and, as is typical for the inspection process, the NRC's inspection/audit scope of the MOV program was expanded. This review consisted of several team inspections and various individual inspections. It should be noted that NYPA implemented measures beyond those recommended by GL 89-10 to enhance the Indian Point 3 MOV program prior to restart. In addition to the MOV dynamic testing conducted during the extended shutdown, GL 89-10 MOV actuators have been rebuilt, MOV pressure locking and thermal binding effects have been assessed, and corrective actions have been implemented where needed. Enclosure 2 to this letter is a copy of the NRC MOV team inspection report which provides further details. It should also be noted that the items listed as open in the enclosed inspection report have since been closed. Based on this review, the NRC staff concluded that NYPA

has satisfactorily determined that Indian Point 3 MOVs subject to GL 89-10 are operable, and that the MOV program is adequate to support operation of the facility.

In your March 29, 1995, letter, you raised a concern regarding the operability of water level instrumentation. As you are aware, the NRC staff has previously evaluated potential errors in level indications following rapid depressurization. All of the affected BWRs have either completed installation of hardware modifications to the water level instrumentation system or are currently in a extended shutdown condition and will install the hardware modifications prior to restart. In addition, the NRC staff has also evaluated potential errors in PWRs on a generic basis and concluded that errors due to non-condensable gases are not significant relative to plant safety. The NRC staff documented the results of these reviews in three safety evaluations dated March 30, 1994, which was forwarded to you on December 18, 1994.

Two special inspection procedures were developed to evaluate the water level issue. Temporary Instruction 2515/119, "Water Level Instrumentation Errors During and After Depressurization Transients (GL 92-04)," was performed at all commercial BWR facilities. Its objective was to verify licensee implementation of operator guidance and training to ensure required operator actions concerning reactor vessel water level following rapid depressurization transients, and also to ensure that this guidance and training is consistent with current plant Emergency Operating Procedures. Temporary Instruction 2515/128, "Plant Hardware Modifications to Reactor Vessel Water Level Instrumentation (NRC Bulletin 93-03)," is being performed at one BWR unit per site, with confirmation of installation at all other affected BWR units. Its objective is to verify and evaluate licensee implementation of hardware modifications to the reactor vessel water level instrumentation by licensees in response to NRC Bulletin 93-03, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRS," and to evaluate the licensee's performance implementing the requirements of 10 CFR 50.59 with respect to this design modification. The NRC continues to conclude that licensees, based on their responses to GL 92-04 and NRC Bulletin 93-03, have provided reasonable assurance that their reactor vessel water level instrumentation is capable of performing its required safety function.

Also in your March 29, 1995, letter, you raised questions concerning the NRC's oversight process and the licensees' obligation regarding condensing pot operability and the conduct of operability determinations. In the NRC's June 22, 1994, response to your 10 CFR 2.206 petition request, the specific concerns regarding operability and operability determinations of condensing pots were addressed. Subsequently, on various occasions you have raised additional questions regarding the generic topic of operability and operability determinations. These include: (1) a May 11, 1995, public meeting between the NRC staff and Boston Edison regarding the Pilgrim Nuclear Power Station; (2) an April 27, 1995, public meeting between the NRC staff and NYPA regarding Indian Point 3; (3) your March 29, 1995, letter to me regarding boiling-water reactor vessel level and pressurized-water reactor pressurizer level condensing pots; (4) your March 16, 1995, INTERNET e-mail to me

regarding operability issues at the Haddam Neck power plant; and, (5) your September 9, 1994, letter to William Raymond of the NRC staff regarding operability issues at Haddam Neck.

The following comments respond to your generic concern of the NRC's oversight process and licensees' obligations regarding operability and operability determinations. The NRC protects public health and safety by assessing licensee compliance with regulatory requirements. This is accomplished, in part, by the inspection process, which audits a licensee's operation in order to determine whether there is compliance with NRC requirements and whether there are any conditions which might undermine reasonable assurance of the protection of the public health and safety. In general, if the inspection process detects a weakness in the area being audited, the audit is expanded so that enough data can be assessed to properly determine whether there is compliance with applicable requirements, or to take enforcement action if a licensee is found to be in noncompliance with NRC requirements.

Operability is the capability of a system and/or a component to perform its specified safety function. The TSs, by virtue of Section 182a of the Atomic Energy Act of 1954, as amended, are part of the NRC-issued operating licenses, and require that systems and/or components important to safety be operable. For example, at Indian Point 3, TSs Section 3.3.A requires an operable safety injection system while the plant is at power. There is, however, no NRC requirement that licensees perform operability determinations per se. Operability is evaluated on a continuing basis by licensees, using various appropriate methods which include day-to-day operations, plant tours, observations from the control room, surveillances, test programs, and other similar activities. As a practical matter, if operability of a system or component comes into question, a licensee is required to ensure that the system can perform its specified safety function.

The relationship of 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to equipment operability also deserves comment. Specifically, Criterion XVI of Appendix B requires, in part, that conditions adverse to quality be promptly identified and corrected. Criterion XVI, does not, per se require performance of operability determinations for any equipment or system when a condition adverse to quality is identified. However, performance of an operability determination may be one appropriate response to a particular condition adverse to quality. In addition, Criterion XVI does not transform the guidance of generic letters, NUREGS, regulatory guides, or other documents into regulatory requirements.

In particular, the NRC issued GL 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Non conforming Conditions and on Operability," in order to inform licensees of NRC internal guidance used during NRC review of licensee operability determinations and of licensee resolutions of degraded and non conforming conditions. The guidance provided by GL 91-18 imposes no requirement upon

NRC licensees, but merely provides information which may be useful to licensees in their efforts to

comply with regulatory requirements. Furthermore, Criterion XVI does not impose any requirement upon licensees to evaluate their operations under GL 91-18. Accordingly, in view of above, the NRC staff will not investigate whether all licensees have conducted operability determinations as requested by your correspondence dated March 16, 1995, and March 29, 1995.

You are correct that my March 9, 1995, letter was in error in stating that the IG investigated the concern in your letter of September 9, 1994, that Mr. William Russell falsely stated that, 'The staff did not review any formal operability determinations,' relating to reactor level condensate pots and that the IG did not substantiate the concern. Although, the IG has conducted an investigation of some of your concerns, the IG has informed us that the concern relating to the alleged false statement has not been investigated and will not be investigated. Further questions or additional information regarding the concern should be directed to the IG.

Finally, your March 29, 1995, letter claims that the NRC refuses to investigate serious violations. This is simply not true. The NRC staff takes very seriously its responsibility of protecting public health and safety. The NRC addresses potential safety issues as they arise, from both a plant specific and generic standpoint. I trust the information provided above serves to clarify the relationship between operability, operability determinations, and NRC requirements.

Please feel free to contact me at (301) 415-1453 if you have any further questions.

Sincerely,

John A. Zwolinski, Deputy Director
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation