October 22, 2003

Mr. David Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1707 H Street NW, Suite 600
Washington, DC 20006

Dear Mr. Lochbaum:

The Petition dated September 8, 2003, as supplemented by letter dated September 22, 2003, submitted by Riverkeeper, Inc. and the Union of Concerned Scientists, and addressed to the Nuclear Regulatory Commission's (NRC's) Executive Director for Operations, Dr. William Travers, has been referred to me pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.206. In the Petition, you requested that the NRC take immediate enforcement action against Entergy Nuclear Operations, Inc. (Entergy), the licensee for Indian Point Nuclear Generating Unit Nos. 2 and 3 (IP2 and 3) in Buchanan, New York. Specifically, you requested that the NRC issue an Order requiring Entergy to immediately shut down IP2 and 3 and maintain the reactors shutdown until the containment sumps are modified to resolve Generic Safety Issue 191 (GSI-191). As an alternative, you requested in the event that the NRC should deny the request to require IP2 and 3 to shutdown immediately, the NRC issue an Order to prevent plant restart following each plant's next refueling outage until such time that the containment sumps are modified to resolve GSI-191. You also requested a requirement to be included within that Order for Entergy to (a) maintain all equipment needed for monitoring leakage of reactor coolant pressure boundary components within containment fully functional and immediately shutdown the affected reactor upon any functional impairment to leakage monitoring equipment, and (b) refrain from any activity under 10 CFR 50.59, 10 CFR 50.90, Section VII.C of the NRC's Enforcement Policy, or Generic Letter 91-18, Revision 1, that increases or could increase the probability of a loss-of-coolant accident (LOCA).

As a basis for your request, you stated your belief that there is a lack of reasonable assurance that the IP2 and 3 containment sumps will be able to perform their function during a LOCA. Your conclusions regarding the containment sumps are based on several publicly available reports that were prepared for the NRC by the Los Alamos National Laboratory (LANL). You cited LANL's findings, documented in NUREG/CR-6762, Volume I, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," (hereafter referred to as the Parametric Study) dated August 2001, as the primary basis for your request to shut down IP2 and 3. You further stated that the requested enforcement actions are appropriate based on prior precedence, including actions taken at the Donald C. Cook and Davis-Besse Nuclear Power Plants in late 1997 and early 2002, respectively.

You met with our Petition Review Board (PRB) on September 24, 2003, to discuss your Petition and provide additional details in support of this request. This meeting was transcribed, and the transcript is publicly available as a supplement to the Petition. After thorough consideration of

your Petition and the information provided during the September 24, 2003, meeting, we are denying your request to immediately shutdown IP2 and 3.

The Parametric Study does not raise sufficient concerns regarding plant specific vulnerabilities of IP2 and 3 to warrant immediate action. We believe that IP2 and 3 are currently operating safely. The Parametric Study was a generic study that did not model individual plants to provide data for drawing conclusions about the operability of a particular sump. The Executive Summary of the Study clearly states that the results are not adequate for that purpose. The Parametric Study does not support the conclusions you have drawn in your Petition regarding the operability of IP2 and 3 sumps. The Parametric Study was specifically designed to answer two questions:

1) Is the emergency core cooling system (ECCS) sump clogging issue a plausible concern for domestic pressurized-water reactors (PWRs)?

2) Is there a need for additional regulatory action regarding PWR sumps?

The Parametric Study answers these questions on a generic, not a plant-specific, basis. It demonstrates that ECCS sump clogging is indeed an issue that merits additional study for PWRs. In order to demonstrate this, LANL conducted a study of 69 cases to determine if there were any typical plant features or characteristics (i.e., plant parameters) that would eliminate sump clogging as a plausible issue for PWRs. The study was conducted using a generic plant piping and containment configuration that was not specific to IP2 and 3. Various plant parameters were then overlayed onto the generic plant. Since randomly overlaying plant parameters onto the generic plant could minimize the applicability of the study results, LANL used combinations of parameters that were reflective of actual licensed plants to determine a reasonable range of sump failure probabilities. Since each case was calculated using a combination of a generic plant piping and containment configuration, some generalized assumptions, and some actual plant characteristics, none of the parametric cases represent any of the 69 operating PWRs. Rather, the study can be used to determine the range of overall possibilities that may exist. Further plant-specific study is needed to assess the sump reliability for individual plants. In any case, this study cannot be used for specific plants such as IP2 and 3.

To better understand the limited applicability of the Parametric Study, it is very important to understand that debris generation and transport are strongly influenced by plant geometry. Factors such as pipe break orientation, locations of debris sources (e.g., different types of insulation, containment coatings, etc.) relative to the break, and locations of plant structures and gratings, all have potentially significant impacts on both the amount of debris generated and the amount transported to the sump. For example, most plants use more than one type of insulation in their containment. In actual cases, some insulation types may only be used in certain locations throughout the containment. Different insulations have significantly different head loss characteristics when entrained onto a sump screen. The Parametric Study lacked sufficient information to model actual debris source locations, so it was assumed that all insulation types were homogeneously mixed throughout the containment. This assumption, while adequate for the purposes of the study, distorts the predicted plant response to different break scenarios.

There are significant reasons that make it inappropriate to apply the Parametric Study results to actual plants. For instance, the study uses plant data that is at least 5-7 years old. Some plants have made significant changes during this time. As a result, there are plant characteristics modeled in the study that do not reflect current plant configurations. For instance, both Indian Point units greatly reduced the amounts of calcium silicate (cal-sil) insulation in their containments when they replaced their steam generators. The new steam generators are insulated with fiberglass insulation. The result is that both plants now have minimal amounts of cal-sil. The parametric cases that you concluded represent IP2 and 3 had approximately 40 percent cal-sil. The parametric cases would, as a result, overestimate head losses relative to the Indian Point plants because cal-sil debris has substantially higher head loss characteristics than fiberglass insulation. This overestimate leads to an inflated risk estimate that is not representative of IP2 and 3.

Another example of changes made at Indian Point Unit 2 is a modification that improves the design of the ECCS low-pressure recirculation pumps. This modification increased the licensing basis net positive suction head (NPSH) margin from 0.97 feet of water (ft H_2O) to approximately 2.5 ft H_2O by decreasing the NPSH required by the pump. This change occurred after the data for the Parametric Study was obtained.

Because the Parametric Study was not intended to draw conclusions regarding specific plants, the information used in the study was not verified with nuclear power plant licensees for accuracy after the study was completed. Entergy has stated that the parametric cases you believe reflect the two Indian Point units utilized ECCS recirculation flow rates that are approximately double each plant’s actual flow rate. Since the pressure drop (i.e., head loss) across a sump screen is directly proportional to the velocity squared, the calculated head losses in the Parametric Study for these two parametric cases would be high by a factor of approximately four.

The Parametric Study used licensing basis NPSH margins as the criteria for determining sump failure. Licensing basis NPSH margins are calculated assuming the pumps are running at maximum flow rates. In the event that recirculation is needed, licensees typically operate their pumps at much lower flow rates. Therefore, NPSH margins calculated at actual expected operating conditions would be substantially (about 2 to 3 times in the case of IP2 and 3) better than the licensing basis calculations.

A very important limitation on the Parametric Study is that it does not model unique plant-specific features. In the case of the Indian Point plants, both units have two sumps in containment; an ECCS sump and a second containment sump that can be used for recirculation. This second sump is located in a different part of the containment from the ECCS sump, utilizes the residual heat removal (RHR) pumps instead of the ECCS pumps, and does not run during an accident unless initiated by operator action (i.e., will not collect debris while the recirculation sump is operating). The RHR pumps have an NPSH margin of approximately 8.4 ft. H_2O when operated at normal flow rates from the containment sump. This back-up system can be used if the normal ECCS recirculation sump loses suction due to debris clogging of the sump screen, and provides an additional layer of safety.

On page 15 of the Petition, you indicate that the Indian Point extra sump has no impact on safety because the Parametric Study already analyzed this feature when they considered phased introduction of recirculation (i.e., half ECCS flow at a time). This statement is incorrect. Utilizing half ECCS flow in a phased manner is modeling the use of one ECCS train at a time. The IP2 and 3 containment sump feature provides an additional sump not considered in the study. It is not another train of ECCS, but an entirely separate system. Because of all of the reasons cited above, the Parametric Study does not provide an adequate basis for drawing conclusions regarding the adequacy of the IP2 and 3 recirculation sumps.

In addition, your use of prior regulatory actions at the Davis-Besse and Donald C. Cook Nuclear Power Plants as a supporting basis for your requested immediate action is not applicable in the case of IP2 and 3. The Donald C. Cook Nuclear Power Plant shutdown voluntarily based on plant-specific information that called into question the adequacy of the sump design. Upon further evaluation, the licensee's engineering staff concluded that the sump design was adequate, and no modifications to the sump were made. The Davis-Besse Nuclear Power Plant was shutdown as a result of the reactor vessel head issue. During this outage, the licensee identified unqualified coatings inside containment. The sump screens were enlarged to resolve this issue. In both cases, the decisions were based on plant-specific information. At this time, given the nature of the Parametric Studies, there is no evidence that IP2 and 3 have a deficiency. As part of the Generic Issue Program, all PWRs, including Indian Point, will perform an evaluation of the potential for debris-clogging based on state-of-the-art methods using plant-specific information.

We agree that GSI-191 is an important issue, and it is currently being addressed through our Generic Issue Program. We have developed and are following through with an action plan for resolving GSI-191. All PWR licensees, including IP2 and 3 have been participating in the resolution process. If our continued studies indicate that unsafe conditions exist at Indian Point or any other plant, we will take immediate actions to ensure the continued health and safety of the public. While many plants, including IP2 and 3, have taken appropriate steps to minimize the risks associated with this issue, an NRC-approved methodology for evaluating each plant's sump performance is being developed to: (1) ensure that each plant evaluates the potential for debris-clogging in a consistent manner based on state-of-the-art methods and plant specific information; and (2) provide the NRC with the technical basis for ensuring that any proposed solution adequately addresses the issue.

We consider continued operation of PWRs during the implementation of the GSI action plan to be safe because: (1) licensees have implemented compensatory measures to mitigate risks associated with the issue. These actions were in response to the NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," issued on June 9, 2003; and (2) the occurrence of any accident, especially one that could potentially challenge the sump, is very unlikely. The more likely accidents (small and medium LOCAs) require less ECCS flow, take more time to use up the water inventory in the refueling water storage tank (RWST), and in some cases may not even require the use of recirculation from the ECCS sump because the plant operators would have sufficient time to safely shut down the plant.

We also note that there are sources of safety margin in PWR designs that may not be credited in the licensing basis for each plant. For instance, NPSH analyses for most PWRs

conservatively calculate NPSH margin using maximum flow rates and do not credit containment overpressure (which may be present during a LOCA). For example, calculating a more realistic NPSH margin at actual expected flow rates and accounting for any containment pressure greater than that assumed in the licensing basis NPSH analysis would demonstrate that there is additional margin for ECCS operability during an accident. Design margins, such as this example, may prevent complete loss of ECCS recirculation flow or increase the time available for operator action (e.g., refilling the RWST) prior to loss of flow. In addition, many plants have plant-specific design features which may minimize potential blockage of the ECCS sumps or may provide other ways to mitigate the sump clogging during a LOCA.

Based on the LANL risk studies (i.e., NUREG/CR-6771, August 2002, that does not account for the impact of potential recovery actions, and the follow on study LA-UR-02-7562, February 2003, that accounts for the impact of potential recovery actions) and using the best and most current information available on pipe-break frequencies (i.e., NUREG/CR-5750), the average plant core damage frequency (CDF) calculated for the GSI-191 containment sump issue is about 5E-6 per year (or one core damage event due to this issue every 200,000 years for each reactor). This estimate indicates that it is safe for plants that implement compensatory measures, such as those that are requested by the Bulletin (e.g., alternate water sources or refilling RWST), to continue to operate while they are performing the necessary plant-specific analyses. The estimate does warrant further plant-specific analyses. If these analyses identify the potential for substantial safety enhancements, they will be promptly implemented.

As provided by Section 2.206, we will fully evaluate the alternative actions you requested and will document the staff’s final decision in a Director’s Decision, which will be available for public comment, within a reasonable time. As you are aware, Brian Benney is the Petition Manager for your Petition. Mr. Benney can be reached at (301) 415-3764. I have enclosed for your information a copy of the notice that is being filed with the Office of the Federal Register for publication. A copy of the Petition, its supplement, and the meeting transcript have been made publicly available on the NRC’s Web site via the NRC's Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room under Accession Nos. ML032580235, ML032760576, and ML032790200 respectively. For your information, you can

find a copy of Management Directive (MD) 8.11 "Review process for 10CFR 2.206 Petitions," at http://www.nrc.gov/reading-rm/doc-collections/petitions-2-206/md08-011.pdf.

Sincerely,

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

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/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

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cc:

Mr. Gary Taylor
Chief Executive Officer
Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. John Herron
Senior Vice President and
Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Fred Dacimo
Vice President, Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 2
P.O. Box 249
Buchanan, NY 10511-0249

Mr. Christopher Schwarz
General Manager, Plant Operations
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 2
P.O. Box 249
Buchanan, NY 10511-0249

Mr. Dan Pace
Vice President Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Randall Edington
Vice President Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John Kelly
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Ms. Charlene Faison
Manager, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. James Comiotes
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 2
P.O. Box 249
Buchanan, NY 10511-0249

Mr. John McCann
Manager, Licensing
Entergy Nuclear Operations, Inc.
Indian Point Energy Center
295 Broadway, Suite 2
P. O. Box 249
Buchanan, NY 10511-0249

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector, Indian Point 2
U. S. Nuclear Regulatory Commission
295 Broadway, Suite 1
P.O. Box 38
Buchanan, NY 10511-0038

cc:

Senior Resident Inspector, Indian Point 3
U. S. Nuclear Regulatory Commission
295 Broadway, Suite 1
P.O. Box 337
Buchanan, NY 10511-0337

Mr. Peter R. Smith, Acting President
New York State Energy, Research, and
Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Mr. J. Spath, Program Director
New York State Energy, Research, and
Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
Electric Division
New York State Department
of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

Mr. Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. Ray Albanese
Executive Chair
Four County Nuclear Safety Committee
Westchester County Fire Training Center
4 Dana Road
Valhalla, NY 10592

Ms. Stacey Lousteau
Treasury Department
Entergy Services, Inc.
639 Loyola Avenue
Mail Stop: L-ENT-15E
New Orleans, LA 70113

Mr. William DiProfio
PWR SRC ConsultaNT
139 Depot Road
East Kingston, NH 03827

Mr. Dan C. Poole
PWR SRC Consultant
20 Captains Cove Road
Inglis, FL 34449

Mr. William T. Russell
PWR SRC Consultant
400 Plantation Lane
Stevensville, MD 21666-3232

Alex Matthiessen
Executive Director
Riverkeeper, Inc.
25 Wing & Wing
Garrison, NY 10524

Paul Leventhal
The Nuclear Control Institute
1000 Connecticut Avenue NW
Suite 410
Washington, DC, 20036

Karl Copeland
Pace Environmental Litigation Clinic
78 No. Broadway
White Plains, NY 10603

Jim Riccio
Greenpeace
702 H Street, NW
Suite 300
Washington, DC 20001

cc:

Mr. Robert D. Snook
Assistant Attorney General
State of Connecticut
55 Elm Street
P.O. Box 120
Hartford, CT 06141-0120

David Lochbaum, Nuclear Safety Engineer
Union of Concerned Scientists
1707 H Street NW, Suite 600
Washington, DC 20006