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Citizens and Scientists for Environmental Solutions

September 8, 2003

William D. Travers, Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

**SUBJECT: INDIAN POINT ENERGY CENTER – PETITION PURSUANT TO 10 CFR 2.206 – PWR CONTAINMENT SUMP FAILURE**

Dear Dr. Travers:

Pursuant to §2.206 of Title 10 of the Code of Federal Regulations, Riverkeeper, Inc. and the Union of Concerned Scientists (UCS) petition the Nuclear Regulatory Commission (NRC) to take the following enforcement action against Entergy Nuclear Northeast (Entergy):

> **Issuance of an Order requiring Entergy to immediately shut down Indian Point Units 2 and 3 and maintain the reactors shutdown until such time that the containment sumps are modified to resolve the Generic Safety Issue 191 (GSI-191) problem.**

This action is warranted based on the current public health hazard posed by the continued operation of the Indian Point reactors without reasonable assurance against containment sump failure and consequential impairment of the reactor core and containment cooling functions. This action is entirely consistent with actions taken by the NRC for the Donald C. Cook and Davis-Besse nuclear plants.

The petitioners cannot fathom a sound reason why an agency chartered to protect public health and safety would not readily grant this warranted request. But if the NRC will not do this necessary and right thing, the petitioners reluctantly ask the NRC to at least do the least-wrong thing by taking the following alternative enforcement action:

> Issuance of an Order requiring Entergy to prevent restart of Indian Point Units 2 and 3 from their next scheduled refueling outages until such time that the containment sumps are modified to resolve the GSI-191 problem,
>
> and
>
> Inclusion within that Order of a requirement for Entergy to (a) maintain all equipment needed for monitoring leak-before-break (LBB) of reactor coolant pressure boundary components within containment fully functional and immediately shutdown the affected reactor upon any functional impairment to said 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 that a loss of coolant accident occurs.

People living around Indian Point would be subjected to unnecessarily high risk, but only until the next scheduled refueling outages If people living around Indian Point must endure – even temporarily – the reactors operating with impaired safety systems, they at least must be protected from Entergy taking any steps that might increase the likelihood of these safety systems being challenged.

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Entergy is aware of the PWR containment sump issue and has informed the NRC that safety at the Indian Point Energy Center is not compromised. For reasons detailed later in this letter, the petitioners are not persuaded by Entergy's assertions and believe the actions requested by this petition are absolutely necessary to responsibly deal with the safety hazard.

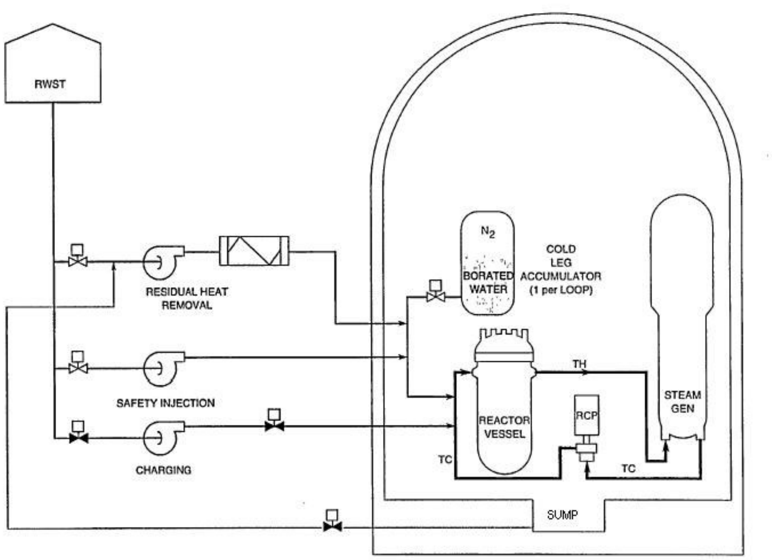
**BACKGROUND ON THE PETITIONERS**

Riverkeeper, Inc. is an independent, member-supported, not-for-profit organization whose mission is to protect the Hudson River and to safeguard the New York City drinking water supply that serves over 9 million New Yorkers. Since its founding in 1966, Riverkeeper has led the battle to restore the Hudson River and has successfully prosecuted hundreds of environmental law breakers. Riverkeeper uses law, science, and education to defend every citizen's right to clean water and healthy communities. Riverkeeper serves as the region's chief environmental watchdog group, acting on hundreds of pollution complaints each year, which are generated through a website (www.riverkeeper.org), telephone hotline, and waterway patrols.

The UCS is a nonprofit partnership of scientists and citizens combining rigorous scientific analysis, innovative policy development, and effective citizen advocacy to achieve practical environmental solutions. The UCS had 61,300 members in 2002.[1] The UCS has actively worked on nuclear power plant safety issues for more than 30 years. When nuclear plant safety levels have been or may be compromised, the UCS engages the Nuclear Regulatory Commission, Congress, local citizens and citizen groups, and the media seeking resolution of the problems. The UCS releases reports, issue briefs, testimony, and presentations which are disseminated by e-mail, handouts at conferences, and via a website (www.ucsusa.org).

**BACKGROUND ON THE PWR CONTAINMENT SUMP PROBLEM**

The Indian Point Energy Center features two operating pressurized water reactors (PWRs) supplied by the Westinghouse Electric Corporation. The PWR gets its name from the fact that water flowing through the nuclear core inside the reactor vessel is maintained under high pressure (approximately 2,200 pounds per square inch) to prevent it from boiling even though it gets heated to over 500° F. The hot water goes through pipes from the reactor vessel to four steam generators located inside the containment building. The hot water flows inside thousands of thin metal tubes within the steam generators. Lower pressure water on the outside of the tubes absorbs heat passing through the tubes' walls and boils producing steam that spins a turbine/generator to make electricity. The water coming out of the steam generator tubes – about 20° F cooler – is pumped back to the reactor vessel to be reheated.




If the reactor vessel gets a hole in it or the piping between the reactor vessel and steam generators breaks, the high pressure forces water out through the opening very rapidly. This is called a loss of coolant accident (LOCA), because the water removes heat produced by the nuclear fuel, thus cooling it. If this heat is not removed, the nuclear fuel will be damaged from overheating.

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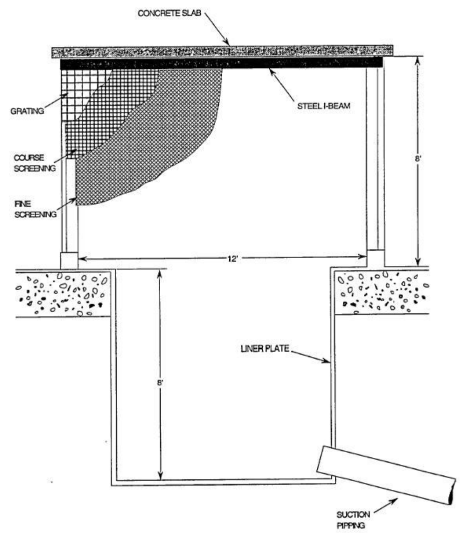
[1] Union of Concerned Scientists, Cambridge, MA, "Annual Report 2002."

When plant sensors detect a LOCA, such as by the rapid drop in pressure inside the reactor vessel, safety features automatically begin to supply makeup water. For example, the charging and safety injection pumps will automatically supply makeup water taken from the refueling water storage tank (RWST).

Even if the RWST were infinitely large or a backup to the RWST was available, at some point the operators must switchover the source of makeup water from the RWST to the containment sump. Recall that the makeup water is needed to compensate for the water pouring out through a broken pipe or open relief valve. This spilled water drains to the basement of the containment where the sump is located. If only outside water is used, the containment would fill up with water, submerging electrical equipment inside containment that must operate. The sheet weight of the rising water also challenges the structural integrity of the containment. So, operators close the valves from the RWST and open the valves from the sump so that the pumps recycle the water inside containment.

The containment sump is more than an open pit. Metal screens cover the sump to prevent debris from fouling the emergency pumps. The configuration of the containment sumps and their protective screens vary from reactor to reactor, but the figure shows a typical arrangement.




The high-pressure water and steam escaping through a broken pipe essentially scours thermal insulation and protective coatings (i.e., paint) off adjacent piping, equipment and structures. After creating debris, the water transports it to the containment sump. The NRC opened Generic Safety Issues 191 (GSI-191) in September 1996 due to concerns that debris could clog the containment sump screens during a LOCA and cause failure of the containment sump function.

Containment sump failure during an accident either prevents or severely impairs the proper functioning of key safety systems needed to keep the reactor core and containment cool. When the reactor core is not adequately cooled, it can overheat and release its radioactive contents into containment. When the containment is not adequately cooled, it can overheat and discharge its radioactive contents to the atmosphere. Plant workers and the public can be harmed if radioactive materials are released to the atmosphere.

**APPLICABILITY OF THE PWR CONTAINMENT SUMP PROBLEM TO INDIAN POINT**

The NRC has issued numerous reports on the PWR containment sump problem in recent years. Often, the identity of specific reactors is masked through the use of numbers (1-69) instead of names.[2] The petitioners unmasked the Indian Point reactors in these NRC reports via the following means:

1. Table 4.5-5, "Comparison of Westinghouse PWR Containments," of NUREG/CR-5640[3] listed key parameters for the Westinghouse-designed PWRs, including Indian Point. Indian Point Units 2 and 3 were the only Westinghouse-designed PWRs with internal diameters of 135 feet and 2,610,000 cubic feet of free volume.

[2] The petitioners don't know why the NRC protects the identity of reactors in this manner, but feel confident that it is not due to national security reasons. Therefore, the petitioners have no qualms about unmasking reactor identities.
[3] P. Lobner, C. Donahoe, C. Cavallin, and M. Rubin, NUREG/CR-5640, "Overview and Comparison of U.S. Commercial Nuclear Power Plants," September 1990.

2. Table A-1(a), “Plant Parameters for Parametric Study,” of a Los Alamos technical letter report[4] indicated that only two reactors had internal diameters of 135 feet and free volumes of 2,610,000 cubic feet: ID Nos. 19 and 47.

3. Table 6, “Containment Spray Set-Point Survey Results,” of NUREG/CR-6770[5] indicated that Indian Point Unit 2 had a containment spray setpoint of 24 pounds per square inch (psig) while Indian Point Unit 3 had a setpoint of 22 psig.

4. Table A-1(a), “Plant Parameters for Parametric Study,” of a Los Alamos technical letter report[6] indicated that the reactor with ID No. 19 had a 22 psig containment spray setpoint while the reactor with ID No. 47 had a 24 psig setpoint.

The petitioners established that Indian Point Unit 2 had ID No. 47 and Indian Point Unit 3 had ID No. 19 in the NRC’s reports. (By equivalent process, the petitioners established that the Davis-Besse reactor had ID No. 69 in the NRC’s reports. Davis-Besse is the only PWR in the US to have faced and fixed the PWR containment sump problem to date.)

The petitioners found the following information in the NRC’s studies:

| Parameter (from [7] unless noted) | Indian Point Unit 2 | Indian Point Unit 3 | Davis-Besse |
|---|---|---|---|
| Sump screen area | 48 $ft^2$ | 36.1 $ft^2$ | 125 $ft^2$ |
| Sump screen mesh size | 0.250 inches | 0.125 inches | 0.25 inches |
| Fibrous insulation in containment | 31.9 % | 36 % | 2 % |
| Reflective metallic insulation in containment | 8.9 % | 10 % | 98 % |
| Micro (Cal-Sil) insulation in containment | 42.2 % | 39.3 % | 0 % |
| Other insulation in containment | 17 % | 14.7 % | 0 % |
| Time to recirculation switchover[8] | 22.12 minutes | 13.3 minutes | 35 minutes |
| Likelihood that sufficient debris will be transported to the containment sump screens for blockage[9] | 100 % for all LOCA cases, both full and half ECCS flow | 100 % for all LOCA cases, both full and half ECCS flow | 0 % for small-break and medium-break LOCAs; 73 – 90% for large-break LOCAs |

[4] D. V. Rao, B. Letellier, C. Shaffer, S. Ashbaugh, and L. Bartlein, Los Alamos National Laboratory, “GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Figure 5-10, August 2001.
[5] D. V. Rao, K. W. Ross, and S. G. Asbaugh, NUREG/CR-6770, “GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences,” August 2002.
[6] D. V. Rao, B. Letellier, C. Shaffer, S. Ashbaugh, and L. Bartlein, Los Alamos National Laboratory, “GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Figure 5-10, August 2001.
[7] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Table 5-1, “Important Parameters that Define Parametric Case Studies,” August 2002.
[8] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Table A-1(a), “Plant Parameters for Parametric Study,” August 2002.
[9] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, “GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance,” Table B-1, “Fraction of Debris-Transport Box Above FTDL Curve,” August 2002.

The petitioners additionally found that:

- for large-break LOCA scenarios 2 and 4, the Indian Point Unit 2 and 3 reactors each have a 99.9 percent chance of containment sump failure (Davis-Besse had a 60 percent chance of failure),
- for large-break LOCA scenario 7, the Indian Point containment sump failure probability drops to merely 99 percent (Davis-Besse had a 40 percent failure chance),
- for medium-break LOCA scenarios 2 and 4, the Indian Point containment sump failure probability is 90 percent (Davis-Besse had a 10 percent failure chance),
- for small-break LOCA (SLOCA) scenarios 5 and 13, the Indian Point containment sump failure probability is 99.9 percent (Davis-Besse had a 40 percent failure chance),
- for SLOCA scenarios 7 and 15, the Indian Point containment sump failure probability is 99 percent (Davis-Besse had a 10 percent failure chance), and
- for SLOCA scenarios 2, 10, and 18, the Indian Point containment sump failure probability is 90 percent (Davis-Besse had a 1 percent failure chance).[10]

The containment sump failure probabilities were defined in this NRC report as follows: [11]

| Failure Value | Description |
|---|---|
| 99.9 % | "The indicated outcome is ALMOST CERTAIN. … Consideration of all identified uncertainties has been made, and none has been found to have a credible effect on the outcome." |
| 99 % | "The indicated outcome is EXTREMELY LIKELY. … Arguments against this position are not supported by either analysis or data." |
| 90 % | "The indicated outcome is LIKELY. … Arguments against this position are apparently flawed." [emphasis in original] |

Thus, the NRC reports indicate that the containment sumps at the Indian Point Energy Center have a very high likelihood of failing (i.e., ALMOST CERTAIN) during a loss of coolant accident and any arguments to the contrary "are apparently flawed."

The NRC reports for Indian Point and Davis-Besse are instructive. Using equally conservative assumptions for the amount of debris generated during a loss of coolant accident, the amount of debris transported to the containment sumps, and the impact on emergency system performance, the NRC reports indicate that Davis-Besse is less likely to encounter sump failure than Indian Point Units 2 and 3. These results were graphically presented in the following charts:[12]

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[10] J. L. Darby, W. Thomas, D. V. Rao, B. C. Letellier, S. G. Ashbaugh, and M. T. Leonard, NUREG/CR-6771, "GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," Attachment C – "Conditional Failure Probabilities for Sump Failure," August 2002.

[11] J. L. Darby, W. Thomas, D. V. Rao, B. C. Letellier, S. G. Ashbaugh, and M. T. Leonard, NUREG/CR-6771, "GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," Table 4.1, "Guidelines for Assigning Conditional Probabilities to Events with "State-of-Knowledge" Uncertainty," August 2002.

[12] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Table 5-1, "Important Parameters that Define Parametric Case Studies," August 2002.

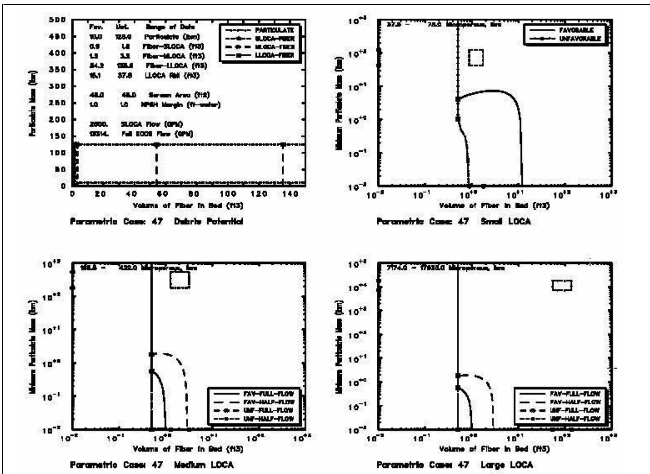


Fig. B-47. Parametric Case 47. Indian Point 2

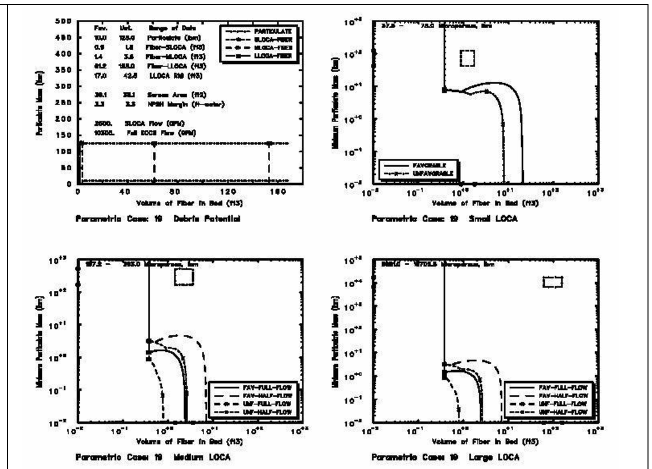


Fig. B-19. Parametric Case 19. Indian Point 3

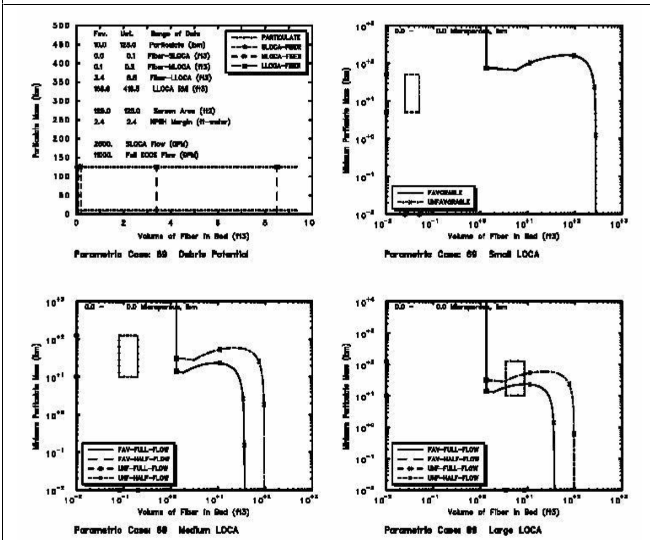
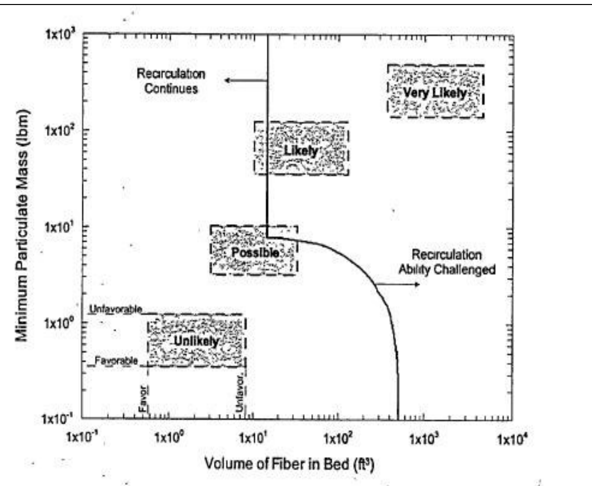


Fig. B-69. Parametric Case 69. Davis-Besse



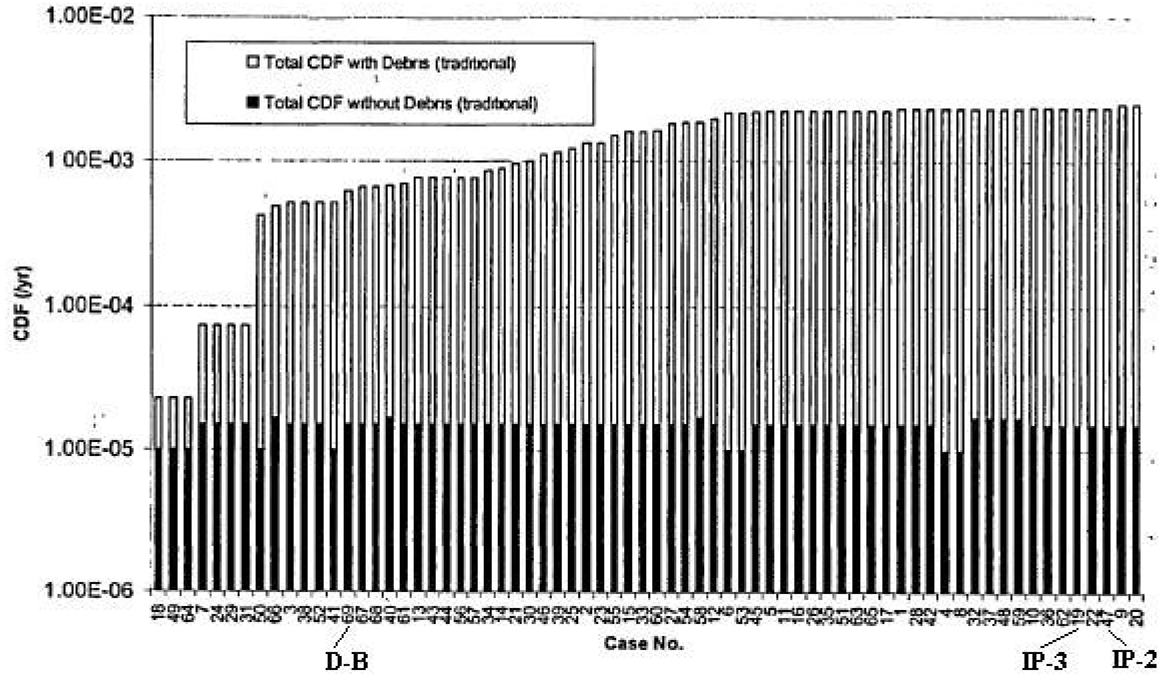

**Examples of Qualitative Assignment of Conditional Sump-Failure Probabilities Developed in "GSI-191 Parametric Evaluations for PWR Recirculation Sump Performance"**

The charts plot the amounts of fiber and particulates calculated to cause failure of the containment sump screens by blockage. The charts may be compared to the amounts of sticks and mud used by beavers to build their dams. The horizontal axis shows the amount of sticks (fiber) needed to hold back the flowing stream. The vertical axis shows the amount of mud (particulates) used. By using mud to fill in the gaps between sticks, beavers can reduce the number of sticks needed to construct the dam. Likewise, the containment sump screens can be blocked by lots of fiber or a lesser amount of fiber in combination with more particulate matter.

Multiple 'blockage' lines are drawn on some charts to account for varied accident responses (i.e., single recirculation pump vs. two recirculation pumps). Superimposed on the charts are boxes indicating the calculated ranges of fiber and particulate matter generated during a loss of coolant accident. As the legend indicates, containment failure is "possible" or "likely" depending on how much of the box straddles the blockage line. When the box is entirely to the left of the blockage line, containment sump failure is "unlikely." Containment sump failure is "very likely" when the entire box is to the right of the blockage line.

At Davis-Besse, the chances of containment sump failure were unlikely except for the large-break loss of coolant accident scenario. For Indian Point Units 2 and 3, the chances of containment sump failure are much, much higher for <u>all</u> loss of coolant accident scenarios.

Containment sump failure impairs the ability to cool the reactor core during an accident, thus increasing the chances of reactor core damage. For Indian Point Units 2 and 3, the containment sump problem increases the core damage frequency (CDF) to $2.368 \times 10^{-3}$ per year.[13] This equates to one core damage event every 422 years for each reactor.




As indicated by the chart from an NRC report,[14] Davis-Besse was among the least-risky plants while Indian Point Units 2 and 3 are among the most-risky plants. Fixing the containment sump at Davis-Besse dropped the risk from the upper value (Total CDF with Debris) to the lower value (Total CDF without Debris), a safety improvement of nearly two orders of magnitude. By NOT fixing the containment sumps on Indian Point Units 2 and 3, the risks remain at the upper value.

There is an industry theory, called leak-before-break (LBB), that a loss of coolant accident will be preceded by some period of increased leakage from the reactor coolant pressure boundary. In other words, pipes won't suddenly burst – as one did at the Surry nuclear plant in Virginia in December 1986 killing four workers – but will leak for a while before breaking. If pipes do leak first and that leakage is detected, operators could shut down the reactor and place it in a safer condition when the pipe finally breaks.

The core damage frequency at Indian Point Units 2 and 3 with the containment sump problem unresolved was determined to be $2.712 \times 10^{-4}$ per year (one accident every 3,687 years) when credit is given for leak-before-break detection and response. [15] Thus, if leak-before-break does occur and if that warning is properly heeded, then the risk of core damage is reduced by nearly a factor of 10.

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[13] J. L. Darby, W. Thomas, D. V. Rao, B. C. Letellier, S. G. Ashbaugh, and M. T. Leonard, NUREG/CR-6771, "GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," page D-3, August 2002.

[14] J. L. Darby, W. Thomas, D. V. Rao, B. C. Letellier, S. G. Ashbaugh, and M. T. Leonard, NUREG/CR-6771, "GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," Attachment C – "Conditional Failure Probabilities for Sump Failure," August 2002.

[15] J. L. Darby, W. Thomas, D. V. Rao, B. C. Letellier, S. G. Ashbaugh, and M. T. Leonard, NUREG/CR-6771, "GSI-191: The Impact of Debris Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," page D-5, August 2002.

The previous owners of the Indian Point reactors calculated a core damage frequency of $3.13 \times 10^{-5}$ for Unit 2 and $4.40 \times 10^{-5}$ for Unit 3.[16] These values correspond to one event every 31,949 years for Unit 2 and one event every 22,727 years for Unit 3. These calculations assumed that the containment sump would not fail and can be viewed as the risk once the containment sump problem is corrected.

In summary, the PWR containment sump problem is applicable to Indian Point Units 2 and 3. It's correction appears even more applicable; hence our petition.

**BASIS FOR THE REQUESTED ACTION**

A recent precedent for the petitioners' preferred requested action (i.e., shutting down both reactors at the Indian Point Energy Center) is the case of the Donald C. Cook nuclear plant in Michigan. The NRC's report to the plant's owner described what happened and why:

> *On September 8, 1997, your staff initiated a dual unit shutdown, and issued a notification of an unusual event (NOUE), as a result of the inability to demonstrate to the team that the ECCS [emergency core cooling system] system would have performed its safety function during post-LOCA conditions under all postulated accident scenarios.*
>
> *Following a postulated LOCA, operation of the ECCS can be divided into two distinct phases, the injection phase, and the recirculation phase. During the injection phase, all ECCS pumps, as well as the containment spray system (CTS) pumps, take their suction from a 350,000 gallon refueling water storage tank (RWST) via branch lines off a common 24-inch suction pipe from the bottom of the RWST. During the postulated LOCA scenario, borated water from the accumulators, reactor coolant system (RCS), and the RWST, combine with the melted borated ice water from the ice condenser (no credit for ice melt water volume assumed in the accident analysis), and is collected in the containment recirculation sump. Near the completion of the LOCA injection phase, when the RWST inventory has been nearly depleted, the suction of the RHR and CTS pumps is manually switched to the containment recirculation sump, beginning the LOCA recirculation phase. … Successful operator switchover to the recirculation phase of ECCS operation following a LOCA is dependent on the containment recirculation sump receiving an adequate amount of water prior to the manual switchover from the RWST. The licensing basis for the plant was a containment water level at an elevation of 602' 10". … The [NRC inspection] team requested a copy of the calculation that established the elevation at the 602' 10" value. However, the licensee could not produce this calculation. … In addition, further licensee evaluation indicated that under certain LOCA scenarios, the volume of water that is in the "active sump" (containment recirculation sump outside of the reactor annulus space) may not be adequate to support long term ECCS or CTS pump operation during the recirculation phase of a LOCA. This conclusion was reached because the licensee could not confirm that sufficient communication (i.e. drainage paths) existed between the active and inactive sumps within the containment, and that water was being removed, over time, from the active containment sump to the inactive sumo via the containment spray system. … Based on these concerns, the licensee declared both trains of ECCS and CTS inoperable in both units, made a notification of an unusual event (NOUE), entered TS 3.0.3, and commenced an orderly dual unit shutdown on September 8, 1997.*[17]

Thus, the two reactors at the Donald C. Cook nuclear plant were shut down because there was insufficient assurance of the ability to establish and maintain emergency core cooling system (ECCS) operation during the recirculation phase of an accident – almost exactly the conditions that exist today at the Indian

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[16] Source: NRC's Individual Plant Examination (IPE) database that was available online until October 2001.

[17] Letter dated November 28, 1997, from Stuart A. Richards, Chief – Events Assessment, Generic Communications, and Special Inspection Branch, Nuclear Regulatory Commission, to E. E. Fitzpatrick, Vice President Nuclear, Indiana Michigan Power Company, "Donald C. Cook, Units 1 & 2 Design Inspection (NRC Inspection Report No. 50-315, 316/97-201)."

Point Energy Center. Neither the plant's owner nor the NRC was certain that the reactors at DC Cook were unsafe. But neither was reasonable certain that the reactors were sufficiently safe, so the responsible thing to do was shut down both reactors.

The NRC took enforcement action earlier this year against the owner of the Davis-Besse nuclear plant in Ohio because that PWR operated without adequate assurance of ECCS performance during the recirculation phase of an accident.[18] The containment sump problem was discovered while the Davis-Besse reactor was already shut down fixing another safety problem. But the NRC would not permit the reactor to restart until the sump problem was corrected, too.[19] The petitioners feel that a safety problem so serious as to prevent a downed reactor from restarting requires operating reactors to shut down.

Davis-Besse has additional relevance to this petition and safety at Indian Point. The NRC reports assumed that the containment sump screen at Davis-Besse had a surface area of 125 square feet.[20] Apparently, closer examination determined the sump screen size was actually only 50 square feet.[21] That smaller screen size is larger than the reported screen size for Indian Point Units 2 (48 $ft^2$) and 3 (36.1 $ft^2$). To assure proper ECCS performance following a loss of coolant accident, the containment sump screen at Davis-Besse was enlarged to approximately 1,200 square feet.[22]

Some might argue that the NRC enforcement action taken against Davis-Besse earlier this year is <u>not</u> applicable to Indian Point because Davis-Besse also had issues with inadequate and unqualified equipment coatings. That difference may be true,[23] but it is irrelevant. If it were relevant, FirstEnergy could have restarted Davis-Besse after having only corrected the equipment coatings problems. After all, the sump screen at Davis-Besse was already larger than those reported at the Indian Point Energy Center. But FirstEnergy enlarged the sump screen by nearly a factor of 25 in addition to resolving the equipment coating problems at Davis-Besse.

The NRC reports assumed that the majority of the debris challenging containment sump capability was created by the violent effects of the broken pipe or whatever caused the loss of coolant accident. In that regard, Davis-Besse appeared far less vulnerable to containment sump failure than Indian Point. Most (98 percent) of the insulation at Davis-Besse was reportedly reflective metallic insulation – material generally considered less conducive to being transported to and blocking the containment sump screens than fibrous insulation. Only about 10 percent of the insulation at Indian Point is reported to be reflective metallic insulation. Less threatening insulation and larger sump screens account for Davis-Besse having only a 60 percent chance of containment sump failure during a large-break LOCA whereas containment sump failure was deemed ALMOST CERTAIN at Indian Point.

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[18] Letter dated July 30, 2003, from John A. Grobe, Chairman – Davis-Besse Oversight Panel, Nuclear Regulatory Commission, to Lew W. Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station NRC Integrated Inspection Report 50-346/2003-015 – Preliminary Yellow Finding."

[19] Letter dated January 28, 2003, from J. E. Dyer, Regional Administrator, Nuclear Regulatory Commission, to Lew Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, "Davis-Besse Oversight Panel Restart Checklist, Revision 2," Item 2.c.1.

[20] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Table 5-1, "Important Parameters that Define Parametric Case Studies," August 2002.

[21] Letter dated August 8, 2003, from Lew W. Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, to Nuclear Regulatory Commission, "Davis-Besse Nuclear Power Station 60-Day Response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors."

[22] Letter dated August 8, 2003, from Lew W. Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, to Nuclear Regulatory Commission, "Davis-Besse Nuclear Power Station 60-Day Response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors."

[23] The petitioners use 'may be true' instead of 'is true' because the NRC has not independently verified Entergy's assertion of containment coating adequacy at Indian Point.

Further evidence that the Indian Point containment sump screens are likely undersized, and therefore vulnerable, comes from the work done to resolve a very similar safety problem on the boiling water reactors (BWRs) in the US. BWR owners faced and fixed the comparable ECCS pump suction strainer issue nearly a decade ago. They enlarged the strainers to reduce the chances that debris prevents proper ECCS pump operation:[24]

| BWR Plant | Total Area per Plant ($ft^2$) |
|---|---|
| Browns Ferry 2 & 3 | 1,192 |
| Brunswick 1 & 2 | 1,575 |
| Clinton | 6,057 |
| Cooper | 2,164 |
| Dresden 2 & 3 | 475 |
| Duane Arnold | 1,359 |
| Fitzpatrick | 2,928 |
| Fermi 2 | 2,322 |
| Grand Gulf | 6,253 |
| Hatch 1 & 2 | 1,110 |
| Hope Creek | 3,788 |
| LaSalle 1 & 2 | 500 |
| Limerick 1 & 2 | 2,715 |
| Monticello | 1,224 |
| Nine Mile Point 1 | 1,286 |
| Nine Mile Point 2 | 1,412 |
| Oyster Creek | 1,425 |
| Peach Bottom 2 & 3 | 3,550 |
| Perry | 5,326 |
| Pilgrim | 1,340 |
| Quad Cities 1 & 2 | 832 |
| River Bend | 2,424 |
| Susquehanna 1 & 2 | 1,340 |
| Vermont Yankee | 2,488 |
| WNP 2 (now Columbia) | 825 |

The smallest BWR strainer, that at Dresden Units 2 and 3, is nearly 10 times larger than the reported screen sizes for Indian Point Units 2 and 3. The typical BWR strainer is about the same size as the enlarged sump screen at Davis-Besse.

In 1997, the NRC provided guidance to nuclear plant owners on the agency's expectations when new information, such as that contained in the NRC reports cited above, becomes available:

> *In the course of its activities, the licensee may discover a previously unanalyzed condition or accident. Upon discovery of an existing but previously unanalyzed condition that significantly compromises plant safety, the licensee shall report that condition in accordance with 10 CFR 50.72 and 50.73, and put the plant in a safe condition.*[25]

[24] D. V. Rao, C. J. Shaffer, M. T. Leonard, and K. W. Ross, NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," Table 8-1, "Total Strainer Area and Vendor for Each BWR Plant Responding to NRCB 96-03," February 2003.

[25] NRC Generic Letter 91-18 Rev. 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," Attachment – Section 2.4, "Nonconforming Conditions," October 8, 1997.

These NRC expectations contain two key elements: (1) discovery of a condition that significantly compromises safety, and (2) placing the plant in a safe condition. That the PWR containment sump problem rises to and exceeds the threshold for "significantly compromising" plant is evident from the NRC reports, which were summarized as follows:

> *NUREG/CR-6771 showed that debris effects resulted in a core damage frequency (CDF) for LOCA events that was almost 140 times the CDF without considering debris when traditional initiating event frequencies are used. (Note: corrections to the NUREG/CR-6771 results are reflected here.) Allowing for leak before break, the CDF with debris was 45 times the CDF without debris. The analysis discussed there shows that, considering the effects of debris and allowing for recovery, the CDF resulting from LOCA events for pressurized water reactors is on average 19 times higher than the CDF when debris effects are not considered. Allowing for leak before break initiating frequencies, the CDF with debris and recovery is twice the CDF without considering debris*.[26]

**A two order of magnitude increase in core damage risk surely constitutes a significant compromise of reactor safety.** Any increase in core damage risk that bumps the value to greater than $1x10^{-5}$ per year is unacceptable; if, that is, the NRC follows its own published guidance on the subject.[27]

That shutdown is the appropriate response to the problem is evident from the Technical Specifications issued by the NRC for the Indian Point reactors. Section 1.0 of the Technical Specifications define operable as follows:

> *A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its intended safety function(s).*[28]

A containment sump that is ALMOST CERTAIN, EXTREMELY LIKELY, or LIKELY to fail in the event it is needed is simply not operable. The Indian Point Technical Specifications are explicit when it comes to the response to inoperable equipment:

> *In the event a Limiting Condition for Operation (LCO) and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least hot shutdown within the next 7 hours, and in at least cold shutdown within the following 30 hours unless corrective measures are completed that restore compliance to the LCO within these time intervals as measured from initial discovery or until the reactor is placed in a condition in which the LCO is not applicable.*[29]

Thus, the immediate (i.e, within 7 hours) shut down of both reactors at Indian Point is the appropriate response to the discovery of a significant compromise to nuclear safety. The conditions today at the Indian Point Energy Center are very comparable to those at the Donald C. Cook nuclear plant in September 1997. There was insufficient assurance of ECCS performance during the recirculation phase of an accident at Cook, so the ECCS functions dependent on the containment sump were declared inoperable and both reactors were shut down. There's insufficient assurance of ECCS performance during the recirculation phase of an accident at Indian Point (given the ALMOST CERTAIN containment sump failure), so both reactors should be shut down. It was the right thing to do then, it remains the right thing to do now.

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[26] K. T. Kern and W. R. Thomas, Los Alamos National Laboratory, LA-UR-02-7562, "The Impact of Recovery From Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency," February 2003.
[27] Regulatory Guide 1.174 Rev. 1, November 2002, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," page 1.174-17.
[28] Indian Point Unit 2, Technical Specification Amendment No. 152, Specification 1.3, 1991.
[29] Indian Point Unit 2, Technical Specification Amendment No. 152, Specification 3.0, 1991.

**BASIS FOR THE ALTERNATIVE REQUESTED ACTION**

If, for some reason, the NRC opts not to do the right thing by granting the first action requested by the petitioners, we believe the least wrong thing for the agency to do is to grant the alternative requested action. This would allow the Indian Point reactors to continue operating until their next scheduled refueling outages, at which time they would be shut down and remain shut down until the containment sump problems were corrected. In the meantime, we'd all hope that a loss of coolant accident did not occur with the containment sump subject to ALMOST CERTAIN failure.

To complement crossed fingers, we petition the agency to take prudent steps to prevent the chances of a loss of coolant accident occurring while the containment sump capability is in question. In other words, recognizing that the ability to mitigate a loss of coolant accident may be impaired to some degree, it's necessary to avoid taking any steps that make such an accident more likely. The NRC reports previously cited indicate that leak-before-break reduces the risk associated with containment sump failure by nearly an order of magnitude (factor of 10). Therefore, the petitioners ask that the NRC order Entergy to maintain all equipment needed for monitoring leak-before-break (LBB) of reactor coolant pressure boundary components within containment fully functional and shutdown the affected reactor within 24 hours of any impairment to said monitoring equipment.

The need to order Entergy to maintain monitoring capability for equipment known to be impaired has precedent at Indian Point. In February 2000, a steam generator tube failed in one of the steam generators on Indian Point Unit 2. There had been warning signs of leaking steam generator tube(s) for weeks prior to this event. But as the NRC team investigating the accident reported:

> *Non-condensable gases are removed from the condensers during normal operation by the SJAEs [steam jet air ejectors] and then released to the atmosphere through the plant vent. The release path is continuously monitored by a radiation detector (R-45). When the activity level reaches the high alarm setpoint, the condenser off-gas flow is automatically diverted to containment.*
>
> *In addition to the condenser off-gas radiation monitors, each of the four MS [main steam] lines is monitored for activity by individual radiation monitors. The monitors are calibrated to detect a specific radioactive isotope (Nitrogen-16) most representative of the primary to secondary leak rate, and provide a reliable method for identifying and trending a SG [steam generator] tube leak during plant operation. The Team noted two previously identified and uncorrected equipment deficiencies that reduced the effectiveness of the MS line radiation monitors for monitoring changes in the pre-event primary-to-secondary leak rate.*
>
> - *The strip chart recorder for the MS line radiation monitors had been out of service since April 1999. The strip chart recorder maintains a continuous recording of the primary to secondary leak rate from all four steam generators. Since April 1999, the licensee has relied on plant chemistry technicians to obtain periodic readings of the leak rate. This equipment problem reduced the pre-vent SG leak rate information that would have been available to the operators*
>
> .
>
> - *The potentiometer used to set the reactor power level input into the leak rate calculation circuit on the MS line radiation monitors has not functioned properly since December 1999. Power level inputs less than the actual reactor power provide a conservative estimate of the primary-to-secondary leak rate while power level inputs greater than actual would provide a non-conservative leak rate estimate. Three condition reports (CRs) have been issued since December 1999 documenting "as found" power level settings ranging from 16% to 108%.*
>
> *The equipment problems discussed above (failed strip chart recorder and malfunctioning power level setpoint potentiometer) degraded the operators ability to monitor changes in the primary-to-secondary leak rate. The resolution of these equipment problems appeared non-timely*

> *considering that the licensee was aware of the degraded condition in the #24 SG prior to the event.*[30]

Thus, in the recent past the Indian Point Unit 2 reactor was operated with known leakage through steam generator tubes, yet its owner allowed the equipment used to monitor such leakage to fall into disrepair. Such irresponsibility was inexcusable then – it must not be repeated now.

Given the important role that detection of leak-before-break can play in lowering core damage risk, it is necessary for the NRC to order Indian Point to maintain optimum monitoring capability and to shut down the reactor when monitoring capability is degraded. The NRC must not meekly assume that this monitoring equipment will be kept functional, particularly at a facility with a track record of "non-timely" resolution of monitoring equipment problems.

To complement the order for optimum monitoring capability, the petitioners ask that the NRC order Entergy to refrain from undertaking any activity that increases the likelihood of a loss of coolant accident until the containment sump problem at Indian Point is corrected. Entergy has several processes available for increasing the likelihood of a loss of coolant accident. They could reduce the scope and/or frequency of inspections of reactor coolant pressure boundary integrity via a license amendment request pursuant to 10 CFR 50.90. They could seek enforcement discretion from complying with licensing requirements pursuant to Section VII.C of the NRC's Enforcement Policy.[31] And they could – without any prior NRC notice and concurrence – approve a temporary change that increases the chances of a LOCA pursuant to 10 CFR 50.59. Absent consideration of the impaired containment sumps, such a temporary change might be justified based on its brevity. But it is unwise to risk it with the reliability of the containment sumps in question. Entergy could – again, without any prior NRC notice and approval – determine pursuant to Generic Letter 91-81 Rev 1[32] to continue operating an Indian Point reactor with degraded reactor coolant pressure boundary pipe supports or indication(s) of pipe cracking or any other threat to reactor coolant pressure boundary integrity. **The NRC should not permit Entergy to undertake any activity that increases the chances of a LOCA until the containment sump problem is corrected.**

**BASIS FOR PETITION VERSUS GENERIC SAFETY ISSUE RESOLUTION PROCESS**

Since September 1996, the NRC has pursued resolution of the containment sump problem for Indian Point and the other 67 affected PWRs in the US via its generic safety issue process. The current schedule calls for GSI-191 to be resolved by March 2007.[33] If the NRC meets its schedule, resolution of this one safety problem will have taken the agency longer than 10 years. By comparison, it took the agency (then called the Atomic Energy Commission) less than 7 years from granting construction permits to issue operating licenses for Indian Point Units 2 and 3.[34] Simply because the NRC is slowly treating this matter generically is insufficient grounds for denying this petition:

> *An unresolved safety issue cannot be disregarded in individual licensing proceedings merely because the issue also has generic applicability; rather, for an applicant to succeed, there must*

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[30] Letter dated April 28, 2000, from Hubert J. Miller, Regional Adminstrator, Nuclear Regulatory Commission, to A. Alan Blind, Vice President – Nuclear Power, Consolidated Edison Company of New York, Inc., "NRC Augmented Inspection Team – Steam Generator Tube Failure – Report No. 05000247/2000-02," Section 4.1.1, "Main Steam Line and Main Condenser Air Ejector High Radiation Monitors."

[31] See http://www.nrc.gov/reading-rm/doc-collections/enforcement/notices/

[32] NRC Generic Letter 91-18 Rev. 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," October 8, 1997.

[33] Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, "Generic Issue Management Control System Report," July 29, 2003.

[34] Nuclear Regulatory Commission NUREG-1350 Vol. 14, "Information Digest: 2002 Edition," Appendix A, "U.S. Commerical Nuclear Power Reactors," June 2002.

*be some explanation why construction or operation can proceed although an overall solution had not been found.*[35]

The petitioners have diligently and in good faith looked for "some explanation why … operation can proceed although an overall solution had not been found" and have not found it. The NRC's reports cited above indicate that containment sump failure at Indian Point in the event of several loss of coolant accident scenarios is ALMOST CERTAIN, to use the words from their conclusion. The rationale for allowing Indian Point to continue running is far less certain.

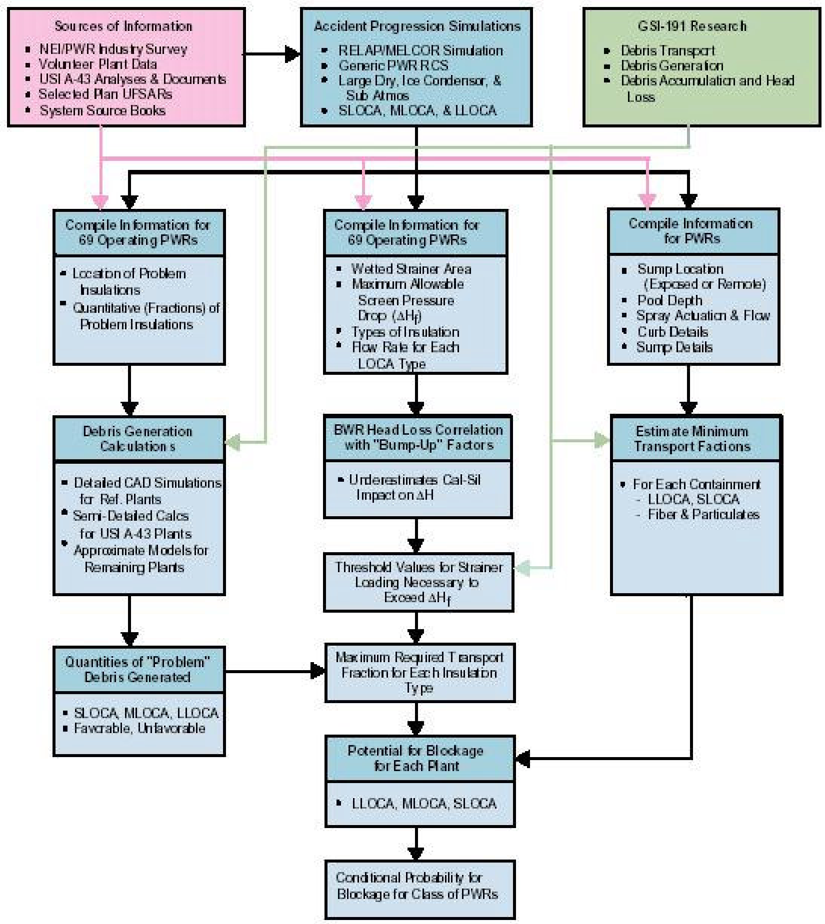


**Fig. 3-2. Technical Methodology Used to Identify Plants Vulnerable to GSI-191 Related Safety Concerns.**

As indicated by Figure 3.2 from an NRC report,[36] the parametric study and other assessments relied on input from the Updated Final Safety Anaysis Reports (UFSARs) and System Source Books (also called Plant Information Books). Prior to 09/11, the petitioners would have been able to access this information and independently determine if the containment sump configuration at the Indian Point Energy Center was as vulnerable as indicated. However, the NRC has removed the UFSARs and Plant Information Books from the public arena – therefore, the petitioners must accept the results from the NRC reports.

The NRC is cognizant of the heightened risk at Indian Point pending resolution of the containment sump problem. In June of this year, the NRC requested that Entergy consider taking compensatory measures at Indian Point to lessen the chances of containment sump failure.[37] The NRC's request was limited to consideration of steps designed to protect the containment sumps <u>after</u> a loss of coolant accident has occurred. The NRC did <u>not</u> request Entergy to consider taking any steps to (a) maintain and/or enhance leak-before-break monitoring capability and (b) minimize the likelihood of a loss of coolant accident. The petitioners seek to avoid events that challenge the containment sumps rather than hope that patchwork measures can cope with their failure during a time of need.

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[35] Nuclear Regulatory Commission, NUREG-0386 Digest No. 10, "United States Nuclear Regulatory Commission Staff Practice and Procedure Digest: Commission, Appeal Board and Licensing Board Decisions July 1972 – December 1998," Section 6.10.2.2, "Effect of Unresolved Generic Issues in Operating License Proceedings," June 2000.

[36] D. V. Rao, B. Letellier, C. Shaffer, S. Ashbaugh, and L. Bartlein, Los Alamos National Laboratory, "GSI-191: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Figure 5-10, August 2001.

[37] Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors," June 9, 2003.

Entergy formally replied to the NRC's request for consideration. Among other things, Entergy stated:

> *IP2 and IP3 are designed with two separate sumps. The recirculation sump, which supplies the recirculation pumps located inside containment, is the normal source for post-switchover core cooling. In the event that adequate core cooling is not established with the recirculation pumps, existing procedures and operator training provide for use of an alternate source consisting of the containment sump, which is also equipped with screens, and the residual heat removal pumps, located outside of containment. The two independent sumps are located on the bottom elevation of containment, but are located approximately 90$^o$ apart. This configuration provides diversity and can be used to reduce the potential for a loss of recirculation capability caused by sump-clogging.*
>
> *Recirculation is initially established by starting only one recirculation pump. … At a later point in the process, a second recirculation pump is started, and the procedure directs operators to maintain the maximum flowrate for both pumps.*[38]

The petitioners are not persuaded by Entergy's response. It was not an explanation for continued operation of Indian Point Units 2 and 3 with impaired containment sumps. First, having two independent sumps is irrelevant when dealing with a common-cause failure mode; namely, blockage of the sump screens by debris generated by the consequences of the loss of coolant accident. Both sumps could be blocked by debris generated by the LOCA and transported by the water. Second, the phased introduction of the recirculation pumps has essentially already been analyzed and found to be insufficient. Los Alamos examined Indian Point for half-ECCS flow (i.e., single recirculation pump) and found that containment sump blockage was LIKELY to ALMOST CERTAIN for all small, medium, and large LOCA scenarios evaluated.[39]

The petitioners are also not persuaded by statements made by the NRC in response to the issue brief on PWR containment sump problems released in August 2003 by the Union of Concerned Scientists.[40] For example, the NRC told reporters for the *Patriot-News* and *The Nation* that the Los Alamos studies were overly conservative. Apparently, the Los Alamos results were meaningful enough to warrant the NRC requiring the owners of every PWR in the US to explore ways to compensate for clogged containment sump screens.[41] The NRC simply cannot have it both ways. If the Los Alamos results are sound enough to require plant owners to compensate for impaired containment sumps, the results are equally sound for the petitioners to use when seeking restoration of prescribed safety levels at the Indian Point Energy Center.

The petitioners are following a precedent established nearly 30 years ago. On October 15, 1973, the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution filed a petition with the Atomic Energy Commission, the forerunner to today's NRC, seeking the shutdown of the Vermont Yankee and Pilgrim nuclear power plants and their sustained shutdown until problems with cracked fuel channel boxes were resolved.[42] Even though "the regulatory staff was aware of the problem, was reviewing it, and was taking action prior to receipt of the petition," the AEC issued a Show Cause Order

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[38] Letter dated August 7, 2003, from Fred Dacimo, Vice President – Operations, Entergy Nuclear Northeast, to Nuclear Regulatory Commission, "60-Day Response to NRC Bulletin 2003-01 Regarding Potential Impact of Debris Blockage of Emergency Sumps."
[39] D. V. Rao, B. C. Letellier, C. Shaffer, S. Ashbaugh, and L. S. Bartlein, NUREG/CR-6762 Vol. 1, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," Table B-1, "Fraction of Debris-Transport Box Above FTDL Curve," August 2002.
[40] Union of Concerned Scientists, "Pressurized Water Reactor Containment Sump Failure," August 20, 2003.
[41] Nuclear Regulatory Commission Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors," June 9, 2003.
[42] Letter dated December 17, 1973, from Voss A. Moore, Assistant Director for Light Water Reactors Group 2, Atomic Energy Commission, to James M. Carroll, Vice President and General Counsel, Boston Edison Company.

to Pilgrim's owner on October 23, 1973, requesting its rationale for continued operation of the reactor.[43] The owner of the Pilgrim nuclear plant responded with letters dated November 14, 1973, and November 30, 1973, contending that the reactor could operate safely at 50 percent of rated power until its next scheduled refueling outage in fall 1974. On December 17, 1973, the AEC's Director of Regulation issued an Order to Pilgrim's owner that mandated:

> *The licensee, not later than during the week commencing December 23, 1973, shut down the Pilgrim Nuclear Power Station and maintain that facility in a safe shutdown condition pending a finding by the Directorate of Licensing that any channel box damage in the facility has been repaired and the cause of any such damaged corrected.*

Thus, the petition process was the proper vehicle for public-interest organizations like the New England Coalition on Nuclear Pollution and the Union of Concerned Scientists to engage the regulator on safety problems at operating nuclear power reactors, even when those problems were already known to and being addressed by the regulatory staff. The 2.206 petition process remains the proper vehicle now for Riverkeeper and the Union of Concerned Scientists to engage the regulator about safety hazards at Indian Point.

If the NRC orders Indian Point Units 2 and 3 to be immediately shut down and it turns out that Entergy's assertions are correct, then the duration of the outage will be limited to the time required for Entergy to submit its justification to the NRC and for the agency to independently verify it. If the NRC orders Indian Point Units 2 and 3 to be immediately shut down and it turns out that Entergy's assertions are incorrect, then the duration of the undue safety hazard to the people living around the plant is limited.

**SUMMARY**

Reports released by the NRC in the past two years indicate there is a very high likelihood that the containment sump will fail during a loss of coolant accident at Indian Point Unit 2 or 3. These reports indicate that the potential problem increases the risk of core damage at Indian Point by two orders of magnitude (factor of 100). Such a risk increase represents an undue threat to public health and safety. Concerns about the containment sump at Davis-Besse prompted the NRC to insist that the reactor not be restarted until after the problem was fixed. The NRC should order Entergy to immediately shut down both reactors at the Indian Point Energy Center and keep them shut down until the containment sump problem is corrected. These serious safety issues must be resolved now, not by March 2007 as NRC currently plans.

If this petition is denied and Entergy subsequently fixes the containment sumps, what will that say about NRC's regard for public health and safety in the interim? If this petition is denied and Indian Point Unit 2 or 3 experiences a LOCA <u>before</u> the containment sumps are fixed, what answers will the NRC have to questions from the victim's families about "Why did you let this happen?" Another Indian Point Lesson Learned Task Force will not be enough.

The NRC's generic safety issue process is an internal one with zero opportunity for formal engagement by the public. Thus, the petitioners must use the 10 CFR 2.206 process because no other process exists to address, and more importantly resolve, this important safety hazard at Indian Point in a timely manner.

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[43] The issue was rendered moot for Vermont Yankee when its owner voluntarily shutdown the reactor in fall 1973 and fixed the fuel channel box problem.

The petitioners request a public meeting with the NRC before the Petitioner Review Board meeting to decide upon the merits of our petition.

If there are any questions, please contact David Lochbaum at (202) 223-6133.

Sincerely,

<ORIGINAL SIGNED BY DAVE LOCHBAUM FOR>

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