Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

REGULATORY MALPRACTICE:
NRC'S "Handling" of the PWR Containment Sump Problem

David Lochbaum, Nuclear Safety Engineer

The Energy Reorganization Act of 1974 created the U.S. Nuclear Regulatory Commission (NRC). This Act of Congress gave the NRC the important, and exclusive, task of protecting the public from the radiological consequences from nuclear power. No other entity at the local, state, or federal level shares this responsibility with the NRC. In fact, all other entities are legally precluded from encroaching on the NRC's regulatory turf. Thus, there is no backup at any level to protect the public when the NRC does its job poorly.

This report exposes regulatory malpractice by the NRC regarding the pressurized water reactor (PWR) containment sump problem. This problem afflicts 68 of the 103 nuclear reactors operating in the United States and makes it much more likely that one of these reactors experiences the ultimate disaster – meltdown with containment failure. This problem has been faced and fixed at literally dozens of PWRs around the world, but only one US PWR has been fixed to date. This report clearly documents that the NRC neither enforces its own regulations adopted to protect the public nor uses the legal means at its disposal to revise regulations it no longer cares to enforce. This report details how the NRC selectively uses risk arguments to enforce or dismiss regulations it sees as convenient. Additionally, this report examines the history of NRC's actions – and inactions – on this safety issue back to the 1970s.

The NRC's regulations are its strength: in general, these regulations are supposed to limit the risk from nuclear reactors to an acceptable level. The NRC's weakness is its inability to consistently enforce its regulations: like the assembly instructions for a bicycle or the directions for medication, regulations are only effective when they are followed. By failing to consistently enforcing its regulations, the NRC is not regulating the risk from nuclear power reactor operations to an acceptable level. The risk is, in fact, unacceptably high by the NRC's own standards.

The purpose of this report is not to chronicle yet another abysmal regulatory failure by the NRC. It certainly achieves that end, but that was not the objective. After all, it could be easily lost in the vast archives of similar reports by UCS, the General Accounting Office, the NRC's Office of the Inspector General, and many others. No, this report seeks to provide the United States Congress with compelling evidence that the NRC is shirking its unique responsibility of protecting public health and safety. It is our hope that the US Congress, which created the NRC with its Energy Reorganization Act, will expeditiously undertake the intensive series of hearings needed to reform the NRC so that it consistently enforces its safety regulations. In other words, we hope the US Congress will do now what it would do in its post-mortem inquiries into a tragic PWR accident – just skipping the part where thousands of Americans get harmed and a large region of our country gets ruined for decades.

Washington Office: 1707 H Street NW Suite 600 • Washington DC 20006-3919 • 202-223-6133 • FAX: 202-223-6162
Cambridge Headquarters: Two Brattle Square • Cambridge MA 02238-9105 • 617-547-5552 • FAX: 617-864-9405
California Office: 2397 Shattuck Avenue Suite 203 • Berkeley CA 94704-1567 • 510-843-1872 • FAX: 510-843-3785

NRC's Failure to Enforce Regulations and Assure Public Safety

Much has been discussed and written about terms like "adequate protection" and "reasonable assurance" as they apply to the regulation of nuclear power plant safety levels. The NRC's Atomic Safety and Licensing Appeal Board (ASLAB) best articulated these concepts, in our view, in this ruling:

> *As a general rule, the Commission's regulations preclude a challenge to applicable regulations in an individual licensing proceeding. … Generally, then, an intervener cannot validly argue on safety grounds that a reactor which meets applicable standards should not be licensed. By the same token, neither the applicant nor the staff should be permitted to challenge applicable regulations, either directly or indirectly. Those parties should not generally be permitted to seek or justify the licensing of a reactor which does not comply with applicable standards. Nor can they avoid compliance by arguing that, although an applicable regulation is not met, the public health and safety will still be protected.* ***For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety***. [emphasis added]
>
> *In short, in order for a facility to be licensed to operate, the applicant must establish that the facility complies with all applicable regulations.* ***If the facility does not comply, or if there has been no showing that it does comply, it may not be licensed***.[1] [emphasis added]

The United States General Accounting Office, among others, acknowledged this ruling:

> *NRC's rules, regulations, and general design criteria (collectively referred to as regulations) are contained in Title 10, Chapter 1, Code of Federal Regulations.* ***NRC regulations are formal legal requirements that utilities must meet to construct and operate their plants***.[2] [emphasis added]

As we understand this ruling, interveners – such as Riverkeeper and ourselves in the recently submitted 2.206 petition on the containment sumps at the Indian Point Energy Center – cannot argue that a reactor meeting all applicable regulations is unsafe, or more legalistically, that a reactor meeting all applicable regulations lacks "reasonable assurance" that the public has "adequate protection." Implicit in the ASLAB's ruling is the obligation on interveners to challenge a regulation they consider insufficiently safe while it is being promulgated. For once the NRC adopts a regulation, it defines "adequate protection" and compliance with it provides "reasonable assurance" of public safety. After its adoption, interveners can only challenge the adequacy of a regulation through the petition for rulemaking process.

As we understand this ruling, licensees – such as Entergy in the recently submitted Indian Point petition – and the NRC cannot argue that a reactor failing to meet all applicable regulations is nevertheless safe. Implicit in the ASLAB's ruling is the obligation on licensees to challenge a regulation considered unnecessarily burdensome while it is being promulgated. After its adoption, licensees can only challenge a regulation through the petition for rulemaking process. Likewise, the NRC cannot choose not to enforce a regulation after its adoption but must change a regulation deemed onerous through a rulemaking proceeding. No matter how well justified, non-compliance with an existing regulation fails to meet the "reasonable assurance" of "adequate protection" standard and violates the regulatory compact between the NRC and the public. When the NRC simply decides not to enforce a regulation, it violates the Administrative Procedures Act by essentially revising a regulatory standard without the legal necessities of public notice and public comment period.

Pre-eminent among many regulations applicable to the pressurized water reactor (PWR) containment sump issue is §50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, of Title 10 in the *Code of Federal Regulations*. This regulation is relatively short and is provided in its entirety in Appendix A.

The Maine Yankee case illustrated how this regulation must be enforced. On December 4, 1995, the NRC received an anonymous letter sent to UCS and forwarded to the State of Maine. The author of the letter alleged that a computer code used to calculate the reactor system's response to postulated accidents had been deliberately manipulated to artificially inflate safety margins. The results from these allegedly 'fudged' analyses had been used to show that operation at an uprated power level of 2,700 Mwt met all requirements of 10 CFR 50.46. On January 3, 1996, the NRC issued an order to Maine Yankee's owner reducing the reactor's maximum output to 2,440 Mwt, the level established by the original operating license.[3] The NRC had not substantiated the allegations before issuing the order, but the agency had concluded that the allegations constituted reasonable doubt about safe operation at 2,700 Mwt. Because the safety analyses for operation at 2,440 Mwt had not been done using the suspect computer code, the allegations did not raise reasonable doubt about operation up to that power level. Thus, the allegations removed the NRC's confidence that Maine Yankee satisfied 10 CFR 50.46 when operated at 2,700 Mwt but did not undermine their confidence that Maine Yankee complied with the regulation at power levels up to 2,440 Mwt. The NRC enforced 10 CFR 50.46 by ordering Maine Yankee to operate only under conditions where reasonable assurance existed.

This regulation, 10 CFR 50.46, is also applicable to the PWR containment sump issue as described by the NRC's Office of Nuclear Reactor Regulation:

> *The staff believes that there is sufficient new information and concerns raised relative to the potential for debris clogging in PWRs that this action plan has been updated to address PWR sump blockage concerns. As noted above, RES's [NRC's Office of Research] parametric evaluation demonstrated that sump blockage is a plausible concern for operating PWRs. The results of the parametric evaluation form a credible technical basis for concluding that sump blockage is a potential generic concern for PWRs; however, the parametric evaluation is ill suited for making a determination that sump blockage will impede or prevent long-term recirculation at a specific plant. Therefore, it is not clear how significant a threat to PWR ECCS operation exists. The staff considers continued operation of PWRs during the implementation of this action plan to be acceptable because the probability of the initiating event (i.e., large break LOCA) is extremely low. More probable (although still low probability) LOCAs (small, intermediate) will generate small quantities of debris, 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 flow through the break would be small enough that the operator will have sufficient time to safely shut the plant down. In addition, all PWRs have received approval by the staff for leak-before-break (LBB) credit on their largest RCS primary coolant piping. While LBB is not acceptable for demonstrating compliance with 10 CFR 50.46, it does demonstrate that LBB-qualified piping is of sufficient toughness that it will most likely leak (even under safe shutdown earthquake conditions) rather than rupture. … And finally, the staff believes that continued operation of PWRs is also acceptable because of PWR design features which may minimize potential blockage of the ECCS sumps during a LOCA. The RES study on sump blockage attempted to capture many of the PWR design features parametrically, however, it is not possible for a generic study of this nature to capture all the variations in plant-specific features that could affect the potential for ECCS sump blockage (e.g., piping layouts, insulation location within containment, etc.). Therefore, evaluation on a plant-specific basis is necessary to determine the potential for ECCS sump clogging in each plant.* [4]

The NRC views the PWR containment sump issue as being a potential threat to Emergency Core Cooling System (ECCS) operation, but claims it lacks the plant-specific analyses needed to make determinations on individual reactors. Rather than make plant owners perform the plant-specific analyses, the NRC had Los Alamos perform pseudo-plant-specific analyses:[*]

> *As part of the GSI-191 study, RES's contractor, Los Alamos National Laboratory (LANL), performed a generic risk assessment to determine how much core damage frequency (CDF) is changed by the findings of the parametric analysis. Utilizing initiating event frequencies that consider LBB credit consistent with NUREG/CR-5750, LANL calculated an overall CDF of 3.3E-06 when debris clogging as a failure mechanism is not considered, and an overall CDF of 1.5E-04 when debris clogging is considered. … The change in CDF is also dominated by the small and very small break LOCAs which are events where there are significant operator actions that can be taken to prevent core damage.* [5]

The two order of magnitude (i.e., factor of 100) risk increase calculated by Los Alamos for the operating fleet of 69 PWRs resulted from their estimates of the likelihood that the containment sump screens would be clogged by debris generated during a loss of coolant accident, thus preventing emergency pumps from supplying cooling water to the reactor. Los Alamos tabulated their estimates for each of 69 PWRs for postulated small, medium, and large loss-of-coolant accidents (SLOCA, MLOCA, and LLOCA respectively). [6] About half of the operating PWRs are "Very Likely" to experience blockage of the containment sumps causing failure of the emergency core cooling system pumps in event of either a small, medium, or large loss-of-coolant accident.

Table 5-7 Results of Parametric Evaluations Regarding Potential for Blockage

ID	SLOCA	MLOCA	LLOCA	ID	SLOCA	MLOCA	LLOCA
1	Likely+	Very Likely+	Very Likely	36	Very Likely+	Very Likely	Very Likely
2	Unlikely	Possible	Very Likely	37	Very Likely	Very Likely	Very Likely
3	Unlikely	Unlikely	Likely	38	Unlikely	Unlikely	Likely
4	Very Likely	Very Likely	Very Likely	39	Unlikely	Possible	Very Likely
5	Very Likely+	Very Likely+	Very Likely	40	Unlikely	Unlikely	Very Likely
6	Likely	Very Likely	Very Likely	41	Unlikely	Unlikely	Likely
7*	Unlikely	Unlikely	Unlikely	42	Likely+	Very Likely	Very Likely
8	Very Likely	Very Likely	Very Likely	43	Unlikely	Unlikely	Very Likely
9	Very Likely	Very Likely	Very Likely	44	Unlikely	Unlikely	Very Likely
10	Very Likely+	Very Likely+	Very Likely	45	Very Likely+	Very Likely+	Very Likely
11	Very Likely+	Very Likely+	Very Likely	46	Unlikely	Possible	Very Likely
12	Possible	Very Likely+	Very Likely	47	Very Likely	Very Likely	Very Likely
13	Unlikely	Unlikely	Very Likely	48	Very Likely	Very Likely	Very Likely
14	Unlikely	Unlikely	Very Likely	49*	Unlikely	Unlikely	Unlikely
15	Unlikely	Likely	Very Likely	50	Unlikely	Unlikely	Possible
16	Very Likely+	Very Likely+	Very Likely	51	Very Likely+	Very Likely+	Very Likely+
17	Very Likely	Very Likely	Very Likely	52	Unlikely	Unlikely	Likely
18*	Unlikely	Unlikely	Unlikely	53	Likely	Very Likely	Very Likely
19	Very Likely	Very Likely	Very Likely	54	Likely+	Likely	Very Likely
20	Very Likely	Very Likely	Very Likely	55	Possible	Likely+	Very Likely
21	Unlikely	Possible	Likely	56	Unlikely	Unlikely	Very Likely
22	Very Likely+	Very Likely	Very Likely	57	Unlikely	Unlikely	Very Likely
23	Unlikely	Possible	Very Likely	58	Very Likely	Very Likely	Very Likely
24*	Unlikely	Unlikely	Unlikely	59	Very Likely	Very Likely	Very Likely
25	Possible+	Possible+	Very Likely	60	Unlikely	Likely	Very Likely
26	Very Likely	Very Likely	Very Likely	61	Unlikely	Unlikely	Likely
27	Likely+	Likely	Very Likely	62	Very Likely+	Very Likely+	Very Likely
28	Likely+	Very Likely	Very Likely	63	Very Likely	Very Likely	Very Likely
29*	Unlikely	Unlikely	Unlikely	64*	Unlikely	Unlikely	Unlikely
30	Possible+	Unlikely	Very Likely	65	Very Likely	Very Likely	Very Likely
31*	Unlikely	Unlikely	Unlikely	66*	Unlikely	Unlikely	Unlikely
32	Very Likely	Very Likely	Very Likely	67	Unlikely	Unlikely	Very Likely+
33	Unlikely	Likely+	Very Likely	68	Unlikely	Unlikely	Very Likely
34	Unlikely	Unlikely	Very Likely+	69	Unlikely	Unlikely	Likely
35	Very Likely+	Very Likely+	Very Likely				

Tally	SLOCA	MLOCA	LLOCA
Very Likely	25	31	53
Likely	7	6	7
Possible	4	6	1
Unlikely	33	26	8

* Zero-Fiber Plant
\+ Ranking Elevated due to Factors Not Considered in Calculations

The NRC's reticence in making plant owners perform the plant-specific analyses to confirm or refute Los Alamos's results is inexplicable. According to paragraph (a)(1)(i) of 10 CFR 50.46:

> ***Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is***

[*] The studies are termed "pseudo-plant-specific" because they relied on plant-specific parameters like containment sump screen size, mesh size, emergency core cooling system flow rates, emergency core cooling system net positive suction head required and available, etc. but apparently lacked something preventing NRC from removing the "pseudo" prefix.

> ***compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded.***

If nothing else, the Los Alamos results raise very serious, credible doubt about the realism in the evaluation models regarding the behavior of the reactor systems during loss-of-coolant accidents. After all, if it is "very likely" that the containment sump will clog, it is also "very likely" that the emergency core coolant system pumps will not be able to supply as much water to the reactor vessel as calculated (and more importantly, as needed to prevent reactor core damage). Paragraph (a)(2) of 10 CFR 50.46 provides the NRC with explicit power to ensure compliance with 10 CFR 50.46:

> ***The Director of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1) (i) and (ii) of this section.***

The Los Alamos results strongly suggested that the evaluations of ECCS cooling performance for most of the operating PWRs were NOT consistent with 10 CFR 50.46. The level of doubt raised by the Los Alamos studies surpasses by far the level of doubt raised by the Maine Yankee computer code allegations. The NRC imposed restrictions on operation at Maine Yankee based on that "reasonable doubt." The NRC has, thus far, essentially ignored substantially greater "reasonable doubt" at the nation's operating PWRs.[#]

But the NRC's reluctance to enforce 10 CFR 50.46 should not have prevented plant owners from complying on their own. The public availability of the August 2001 Los Alamos study should have prompted each PWR owner to estimate the effect of the error in the evaluation model to satisfy 10 CFR 50.46 paragraph (3)(i) which states:

> ***Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.***

Los Alamos estimated that the core damage frequency increased by a factor of 100 for the average PWR due to the containment sump issue. The reactor core cannot be damaged unless the peak fuel cladding temperature difference exceeds 50°F, by bunches. 10 CFR 50.46 paragraph (3)(ii) specifies:

> ***If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with §50.46 requirements.***

[#] The NRC contends it lacks "absolute proof" that any of the 68 PWRs fails to comply with 10 CFR 50.46 and is unable to act absent that proof. But the NRC did not require "absolute proof" in the Maine Yankee case and doesn't need it now. Considering that compliance with 10 CFR 50.46 does not provide "absolute assurance" of public safety, only "reasonable assurance," it is ludicrous that the NRC would require "absolute" proof of a violation rather than "reasonable" proof. By doing so, the NRC affords its licensees greater protection than the public. The public only gets "reasonable assurance" against a nuclear disaster. The licensees can scoff at regulations <u>unless</u> the NRC finds "absolute" proof that it is happening.

Thus, owners of PWRs determined by Los Alamos to be "Very Likely" to experience containment sump blockage and consequential emergency core cooling system pump failure should have evaluated the impact on their 10 CFR 50.46 analyses and reported to the NRC, within 30 days, their proposed schedule for compensatory actions. The PWR owners are not complying with 10 CFR 50.46. The NRC is not enforcing compliance with 10 CFR 50.46. And the public is not getting "reasonable assurance" of "adequate protection."

Nearly two decades ago, the NRC revised the criteria it would use to determine if the containment sump design in the next PWR to be built complied with 10 CFR 50.46:

> *The current 50% screen blockage assumption*[♣] *identified in Regulatory Guide (RG) 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," should be replaced with a more comprehensive requirement to assess debris effects on a plant-specific basis.* ***The 50% screen blockage assumption does not require a plant-specific evaluation of the debris-blockage potential and usually will result in a non-conservative analysis for screen blockage effects****.*
>
> *The staff has revised Regulatory Guide (RG) 1.82, Revision 0, "Sumps for Emergency Core Cooling and Containment Spray Systems" and the Standard Review Plan Section 6.2.2, "Containment Heat Removal Systems" based on the above technical findings….* ***The revised guidance will be used on Construction Permit Applications, Preliminary Design Approval (PDA) applications, and applications for licenses to manufacture that are docketed after six (6) months following issuance of RG 1.82, Revision 1, and Final Design Approval (FDA) applications, for standardized designs which are intended for referencing in future Construction Permit Applications, that have not received approval at six (6) months following issuance of the RG 1.82, Revision 1****.* [7] *[emphasis added]*

Assuming it follows its own revised guidance, the NRC will never again license a pressurized water reactor to operate without a plant-specific evaluation of the debris-blockage potential for the containment sump screens. In revising Regulatory Guide 1.82, the NRC implicitly conceded that the process it had used to show compliance with 10 CFR 50.46 for the nation's operating PWRs was faulted. Yet the NRC failed then, and has failed since then, to enforce another of its regulations, 10 CFR 50.100, Revocation, suspension, modification of licenses and construction permits for cause, of Title 10 in the *Code of Federal Regulations* which states:

> ***A license or construction permit may be revoked, suspended, or modified, in whole or in part,*** *for any material false statement in the application for license or in the supplemental or other statement of fact required of the applicant; or* ***because of conditions revealed by the application for license or statement of fact or any report, record, inspection, or other means, <u>which would warrant the Commission to refuse to grant a license on an original application</u>*** *(other than those relating to §§50.51, 50.42(a), and 50.43(b) of this part); or for failure to construct or operate a facility in accordance with the terms of the construction permit or license, provided that failure to make timely completion of the proposed construction or alteration of a facility under a construction permit shall be governed by the provisions of §50.55(b); or for violation of, or failure to observe, any of the terms and provisions of the act, regulations, license, permit, or order of the Commission. [emphasis added}*

[♣] Basically, this assumption had plant owners calculate how much water their emergency pumps could get from flow through the unblocked half of the containment sump screens. There was no effort by plant owners or NRC to verify that the containment sump screens would not get blocked more than 50 percent.

By its own admission, the NRC would refuse to grant an operating license to any future PWR applicant who had submitted a design based on the old 50 percent containment sump screen blockage assumption in Regulatory Guide 1.82 Revision 0. But the NRC opted not to revoke, suspend, or modify the operating license for any of the PWRs it had already licensed using that flawed basis. Why not? Robert Ripley would not believe this one:

> *However,* ***the staff's regulatory analysis*** *(NUREG-0869, Revision 1, "USI A-43 Regulatory Analysis")* ***evaluated (1) containment designs and their survivability should loss of recirculation occur, (2) alternate means to remove decay heat, (3) release consequences (which were based on pipe break probabilities which did not incorporate insights gained from recent pipe fracture mechanics analyses)****, and (4) cost estimates for backfits considered (i.e., reinsulating). This regulatory analysis did not support a generic backfit action and resulted in the decision that this revised regulatory guidance will not be applied to any plant now licensed to operate or that is under construction.* [8] *[emphasis added]*

The NRC relied on pseudo-plant-specific analyses in deciding NOT to fix the PWR containment sump problem in 1985. Apparently, the NRC viewed the results from such analyses good enough then to justify taking no action. It's curious that the NRC now views the results from pseudo-plant-specific analyses performed by Los Alamos showing that the PWR containment sumps must be fixed as not being good enough to justify taking immediate action. Pseudo-plant-specific analyses have the same merit in 2002 as they did back in 1985. Yet the NRC will reject analytical results showing there is a safety problem and embrace analytical results, of equivalent quality, indicating the lack of a problem. This is a shameful practice on the NRC's part that needs to be eliminated.

The pseudo-plant-specific analyses performed by Los Alamos concluded that the containment sump problem posed a serious safety hazard at the majority of the nation's operating PWRs. The only genuine plant-specific analysis, to our knowledge, was performed for the Davis-Besse nuclear plant:

> *The [Davis-Besse] licensee reported in LER 346/2002-005 deficiencies with the containment emergency sump to perform its function under certain accident scenarios due to clogging of the emergency core cooling and containment spray systems' sump screen by fibrous materials, unqualified coatings, and various other debris.*
>
> *The licensee performed comprehensive and extensive corrective actions. The actions included installation of a larger emergency sump strainer, field walk-downs of potential debris which could potentially clog the emergency sump, and evaluations on potential debris left in containment.* ***Additionally the licensee has analyzed the emergency core cooling system and containment spray system recirculation functions with respect to the potentially adverse post-accident debris blockage effects to confirm compliance with 10 CFR 50.46(b)(5) and all other existing applicable regulatory requirements****.*
>
> ***The NRC conducted inspections and evaluations to verify that the containment sump deficiencies had been adequately resolved****. The NRC reviewed the licensee's corrective actions, including the design modification for the new sump, field installation, and compliance with regulatory requirements. The NRC inspections are documented in Inspection Report 50/346-03-06 and in this Inspection Report.*
>
> ***In September 2, 2003, the Davis-Besse Oversight Panel met to discuss this issue and concluded that Restart Checklist Item 2.c.1 is closed****.* [9] [emphasis added]

These recent events at Davis-Besse confirm what Los Alamos suspected – that the PWR containment sump is vulnerable to clogging with consequential impairment of the reactor core cooling and containment cooling systems. In fact, Los Alamos had determined Davis-Besse to be one of the PWRs least susceptible to the containment sump problem, but the problem at Davis-Besse was very real and had serious safety consequences – the NRC issued a YELLOW finding[**] to FirstEnergy for Davis-Besse containment sump issue.[10] The events at Davis-Besse demonstrate beyond any doubt that plant owners <u>can</u> analyze debris generation, debris transport, debris loading on containment sump screens, and resulting impact on safety system performance and that the NRC <u>can</u> review that analysis and determine if it complies with 10 CFR 50.46. Since it is undeniable that the plant-specific analyses <u>can</u> be performed and reviewed today, there are no legal grounds for continuing to postpone these analyses. This work <u>can</u> be done – the NRC is meekly tolerating it not being done.

The NRC is simply <u>not</u> enforcing 10 CFR 50.46 with regard to the PWR containment sump issue. Enforcing compliance with this federal regulation is the only legal way for the NRC to properly discharge its responsibility to protect public heath and safety:

> ***For, once a regulation is adopted, the standards it embodies represent the Commission's definition of what is required to protect the public health and safety.*** [11] *[emphasis added]*

It is ironic that the NRC is guilty of the same root cause that it assigned to Northeast Utilities' failings at the Millstone Nuclear Power Station:

> ***While there is a strong emphasis on safety as a stated objective, the organization does not consistently recognize or emphasize the collective set of administrative*** *(e.g., the proposed Determine Course of Action (DCA) concept)* ***and technical processes*** *(e.g., Setpoint Control)* ***that demonstrate and assure that objective is met***. [12]

It is unbelievable that the NRC is also guilty of the same sin that its own Special Inquiry Group (also known as the Rogovin committee) identified as a significant contributing factor to the Three Mile Island accident:

> *[C]ategorization of an issue as generic typically delays its resolution.* ***Because issues are regarded on a generic basis and are not regarded as an impediment to individual plant licensing, little incentive exists for their resolution***. [13]

The NRC should be ashamed that it cannot find incentive for resolving generic safety hazards, such as the PWR containment sump problem, in assuring adequate protection of millions of Americans. Instead, the NRC gambles, once again, with the lives of Americans:

> *The Davis-Besse plant, located in Oak Harbor, Ohio, received an NRC operating license in 1977. In 1979 NRC inspected the plant and recommended that the utility install a third auxiliary feedwater pump to correct a design deficiency that NRC concluded could contribute to a core melt during an accident. … However, NRC allowed the utility time to analyze alternatives such as upgrading procedures and control systems before taking the required corrective action. In 1984, 5 years later, the utility agreed to install the third pump by late 1985. However, before the pump was installed, the plant experienced a series of equipment failures and operator errors in June 1985 such that (1) one main feedwater pump became inoperable, (2) the utility could not activate*

[**] Beginning in April 2000, the NRC employs a four-color rating scheme to safety findings – GREEN, WHITE, YELLOW, and RED in order of increasing significance.

> *another feedwater pump, and (3) the auxiliary feedwater system became inoperable.* ***In October 1985 and May 1986 hearings before the Subcommittee on Energy Conservation and Power, House Committee on Energy and Commerce, NRC admitted that although the equipment problems posed an undue risk to public health and safety, the agency waited too long to require the utility to install the third pump.*** *NRC also admitted that its inspection and enforcement program failed to identify the potential for the extensive equipment failures that subsequently occurred, even though the plant's operating performance had declined since 1982.* [14] [emphasis added]

The NRC consistently asserts public health and safety is its top priority, consistently promulgates regulations to provide adequate protection of public health and safety, and inconsistently enforces those regulations. The NRC allows nuclear reactors to routinely operate in violation of the regulations specifically adopted to provide "reasonable assurance" the public has "adequately protection." Thus, the NRC is failing to demonstrate and assure that its objective is being met. It is waiting too long to resolve the PWR containment sump issue. It is exposing millions of Americans to higher risk than is necessary.

Improper Reliance on Probability Arguments

Following receipt of a 2.206 petition on the containment sump problem at the Indian Point Energy Center,[15] NRC staffers have been quoted by the media asserting that the low likelihood of an accident requiring the containment sumps to "save the day" justifies the long-term resolution plan for the issue. At least one NRC staffer was quoted as saying that there's never been a prior event at any PWR where the containment sump was challenged. UCS fails to see the relevance of these NRC statements. Our version of 10 CFR Part 50 doesn't contain footnotes or other caveats explaining that the regulations are only applicable after the first accident. Best we can tell, the regulations are intended to protect the American public from even the first accident.

Earlier this year, the NRC issued an order to every licensee (i.e., all the PWRs and all the BWRs) mandating immediate actions be taken to upgrade security measures:

> *As part of the Commission's review of the security and safeguards program, the Commission has assessed information provided by the intelligence community and determined that revisions to the DBT [design basis threat], as currently specified in Title 10 of the Code of Federal Regulations, §73.1(a), are required.* ***The Commission has determined that the current threat environment requires that the enclosed Order be effective immediately****.* [16] [emphasis added]

No US nuclear plant has ever experienced an event where security was challenged by a terrorist. No publicly available risk assessment suggests, yet along demonstrates, that the terrorist threat is anywhere close to the core damage risks estimated by Los Alamos for the PWR containment sump issue. No publicly available threat assessment suggests that the probability of terrorist attack is nearly as high as the probability of a loss of coolant accident at a PWR. Yet the NRC took action on the security issue and inaction on the PWR containment sump problem.

The NRC is arbitrarily and capriciously using probability in its regulatory decision-making. When it received bad press, the NRC set aside the probability numbers and ordered plant owners to immediately fix security problems. Until it receives bad press, the NRC invokes the probability numbers to postpone fixes of the PWR containment sump problems. The NRC should be more concerned about public safety than its Neilsen ratings.

Containment Sump Issue History

Generic Safety Issue No. 191 (GSI-191) is merely the latest incarnation of the NRC's process for NOT resolving the PWR containment sump problem that has plagued US nuclear plants for over two decades. NRC Chairman Joseph Hendrie wrote to President Jimmy Carter on Valentine's Day 1979 about it:

> *Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing through the break would be collected in the emergency sump at the low point in the containment. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or breaking of the containment.*
>
> *One postulated means of losing the ability to draw water from the emergency sump would be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a pipe break, the subsequent violent release of the high pressure water could rip off the insulation in the area of the break. This debris could then be swept into the sumps potentially causing damage.* [17]

At the time NRC apprised President Carter of a problem that could lead to reactor meltdown, dozens of PWRs were under construction in the United States including many of those operating today. Rather than ensure that each, or any, of these reactors corrected the known safety problem before it commenced operating, the NRC treated the problem as an Unresolved Safety Issue, specifically Issue A-43:

> *Unresolved Safety Issue (USI) A-43 deals with safety concerns related to the availability of adequate recirculation cooling water following a loss-of-coolant accident (LOCA) when long-term recirculation cooling must be maintained to prevent core melt. These safety concerns can be summarized by the following question:*
>
> > *In the recirculation mode, will the sump design (for pressurized water reactors, PWRs) or the residual heat removal (RHR) suction intakes (for boiling water reactors, BWRs) provide sufficient water to the RHR pumps, and will this water be sufficiently free of LOCA-generated debris and air ingestion so as not to impair pump performance, thereby providing adequate net positive suction head (NPSH) to the recirculation pumps?*
>
> *Generic plant insulation surveys, development of debris estimation and transport methods, experiments, and public comments received have shown that debris effects are dependent on the type and quantities of insulation employed, the layout of the containment, and the post-LOCA recirculation requirements.* ***A single, or overall generic, solution is not possible****.* [18] [emphasis added]

Given that the NRC's efforts between 1979 and 1984 concluded that a generic solution was not possible, one wonders why the NRC opened Generic Safety Issue No. 191 in September 1996 to deal with the PWR containment sump problem. After all, using the generic safety issue process to resolve a safety hazard for which you have already determined "*A single, or overall generic, solution is not possible*" seems like a bureaucratic stall tactic. If so, it has worked exceedingly well. If not, the NRC's glacial pace on GSI-191 has reaffirmed their conclusions from the 1980s.

Before the NRC closed USI A-43, they researched the problem and outlined three possible solutions:

> *Industry impact as a result of the proposed actions will vary from plant to plant because of the variations in insulation employed, actual sump design and operational requirements (e.g., NPSH margins must be maintained for the recirculation pumps). Thus the impacts can be categorized as:*
>
> *(1)* ***analysis*** *only required*
> *(2)* ***analysis*** *plus minimal plant backfits required (e.g., enlargement of screen area to trap debris, reduction of recirculation flow rates from current ratings, etc.)*
> (3) ***analysis*** *plus replacement of large quantities of the problem insulation required* [19] *[emphasis added]*

<u>All</u> of the recommended options involved at least a plant-specific analysis. The NRC also looked at the option of doing nothing, but found ample reason to rule it out:

> ***Ignoring the implications of the results of the USI A-43 debris-blockage effects study with respect to OLs [operating licenses] and NTOLs [near-term operating licenses] is not acceptable****. Sump failure probabilities of* $3x10^{-6}$ *to* 10^{-4}*/Rx-yr have been calculated; the results are very strongly plant dependent.* ***The uncertainties associated with types and quantities of insulation utilized (and given the fact that there have been unreported changes in the types of insulation used) warrant followup action****.* [20] *[emphasis added]*

Unless obfuscation and foot-dragging count, the NRC did not do any follow-up action. The NRC closed USI A-43 in 1985 without requiring any operating PWR to fix, or even analyze, anything. [21] The NRC somehow accepted that which it had determined was "not acceptable." The do-nothing option, which had been specifically excised from the ballot, somehow won the election:

> *…****the staff has concluded that no new requirements need be imposed on licensees and construction permit holders****…* [22] [emphasis added]

The NRC, which professes to have public safety as its foremost priority, fabricated Unresolved Safety Issues for the express purpose of allowing nuclear power reactors to start up <u>without</u> fixing known safety problems. Mr. J. Montgomery of the NRC staff explained this gaming system to the Commissioners:

> *One thing I wanted to explain that one of the limitation that we discussed here is a limitation applied to consideration of certain kinds of costs in dealing with issues that were left unresolved at the construction permit stage. That is a result, as Jim has pointed out, of a rather archane legal system but* ***that system was developed deliberately as a accommodation to the nuclear power industry. It was developed in order to get plants licensed without resolving significant safety questions as in the PRDC case. In effect, a kind of deal was struck very early in the industry****.* [23] [emphasis added]

Thus, the NRC intentionally and deliberately avoided protecting the public and instead concentrated on merely getting the nuclear power plants up and running. The NRC has admitted to placing production ahead of safety.

Accommodating the industry at the expense of the public did not stop when the reactors started. A decade later, NRC Chairman Ivan Selin expressed his disappointment over the ongoing accommodation game:

> *…a related area in which the industry must do better is in anticipating generic problems and in solving them early. This need will become more acute as the universe of regulated reactors gets older and new generic aging issues emerge. …* ***When confronted with the problem, the industry's response was to deny its existence without investigation, forcing the NRC to spend much time and resources to prove the problem's existence****. … Later, when the NRC was able to prove that its concern was valid, both of us found ourselves in a position where a safety issue had been known for several years, but corrective action had not yet been taken. …* ***When generic problems such as these are not promptly and fully addressed, both the NRC and the industry find themselves under justifiable criticism****. Additionally, unnecessary financial and organizational resources are often required to deal effectively with such long-festering problems*.[24] [emphasis added]

Chairman Selin was referring to the industry's response, or non-response, to the motor-operated valve problem (i.e., Generic Letter 89-10), but his words apply equally to the PWR containment sump problem and a host of other safety issues. Why was the NRC Chairman even directing his staff "*to spend much time and resources to prove the problem's existence*"? As the NRC points out:

> "*…it is important to recognize that although the staff's review of an application is partially an "audit" review,* ***the applicant for a license is obligated to assure compliance with applicable regulatory requirements. It is the applicant who bears the burden of proof on the issue***." [25] [emphasis added]

Why has NRC usurped the obligations from its licensees? Why has NRC stolen the burden of proof from its licensees? An NRC manager posed this question before the US House of Representatives:

> *My reaction was, "****Who the hell is regulating who?****"* ***In my opinion, the NRC did not have to buckle to industry pressure****.* [26] [emphasis added]

Likewise, the NRC buckled to industry pressure and closed USI A-43 without taking any action to resolve the safety issue at any operating PWR. The NRC claimed it needed plant-specific analyses to define what corrective actions were needed at what reactors. Okay, fair enough. But how could the NRC dismiss the issue without having performed a single plant-specific analysis? If a plant-specific analysis was necessary to definitely demonstrate presence of a safety threat, wouldn't a plant-specific analysis also be required to definitively demonstrate absence of that safety threat? Of course, but that wouldn't accommodate the nuclear industry. UCS hereby accepts Chairman Selin's implied invitation to justifiably criticize the NRC for not promptly and fully addressing the PWR containment sump issue since its inception in 1979: **the NRC's performance on this important safety matter is atrocious**.

Following the Three Mile Island meltdown in 1979, several task forces were empanelled to study the accident and its causes. President Jimmy Carter appointed Dartmouth College President John G. Kemeny to chair one of the task forces. Kemeny's committee concluded:

> *It [the NRC] was clearly not part of the solution but a serious part of the problem. …* ***They had the lovely habit of giving some very difficult issues a special "generic" label, thus allowing these issues to sit on the shelf****.* [27] *[emphasis added]*

A student in school cannot hope to avoid missing questions on an examination by labeling them "generic" and thus unanswerable. A taxpayer in an Internal Revenue Service audit cannot hope to explain questionable deductions by labeling them "generic" donations and thus receipt-free. But nuclear plant owners know they can avoid fixing safety problems by labeling them "generic" and forcing NRC to play the accommodation game. By playing along, the NRC is aiding and abetting practices subjecting millions of Americans to unnecessary risk.

The NRC's irrational behavior is best explained by someone who served on the Nuclear Regulatory Commission at the time when the PWR containment sump problem was first buried. Commissioner James Asselstine bemoaned and protested the NRC's intentional and deliberate abandonment of its mission to protect public health and safety:

> *I can think of no other instance in which a regulatory agency has been so eager to stymie its own ability to carry out its responsibilities. Indeed, the adoption of this [backfititng] rule* ***is the most compelling evidence to date of the Commission majority's open hostility to the regulatory mission of this agency****. …* ***By this step, the Commission is moving in the wrong direction – a direction that will likely result in further serious operating events, more accidents, and a lower level of safety than that achieved in many more forward-thinking countries in the world****.* [28] [emphasis added]

Lest Commissioner Asselstine's remarks are misconstrued as "sour grapes" whining from someone on the short end of a Commission vote, let's examine what "more forward-thinking countries in the world" have already DONE about the PWR containment sump problem: [29]

Country	Completed Actions from 1992 to 2002
Belgium	6-fold increase in containment sump screen areas completed at 2 of 7 operating PWRs
Canada	13-fold increase in containment sump screen areas completed at 4 of 18 operating PWRs; schedules for similar modifications at 10 other PWRs
Czech Republic	4-fold increase in containment sump strainer area completed at 4 of 6 operating PWRs
Finland	Approximate 10-fold increase in containment sump strainer area completed at 2 of 2 operating PWRs
Hungary	Approximate 50-fold increase in containment sump strainer area completed at 4 of 4 operating PWRs
Japan	Approximately 95% of the fibrous insulation replaced by non-fiber insulation at 23 of 23 operating PWRs
Netherlands	50 percent increase in containment sump strainer area completed at 1 of 1 operating PWR
Russia	Significantly larger containment sump strainers installed at 4 of 6 operating PWRs
Slovak Republic	Significantly larger containment sump strainers installed at 4 of 4 operating PWRs

So, while the NRC accommodates, the nuclear regulatory bodies in forward-thinking countries fix! At least 25 PWRs outside the US have resolved the containment sump problem by increasing the physical size of their sump screens to make them less vulnerable to clogging. At least 23 PWRs outside the US have resolved the containment sump problem by replacing insulation inside containment with a type more resistant to becoming debris in event of an accident. Only 1 PWR in the US has resolved the containment sump problem. The remaining 68 PWRs in the US are hiding behind the NRC's shield, which is supposed to protect the public but instead is being misused to prevent the public from getting the "adequate protection" it so richly deserves.

Conclusion and Recommendations

Millions of Americans are exposed to higher risk than necessary because the NRC is not enforcing federal safety regulations. The regulations are the NRC's only legal means for determining with reasonable assurance that the public is adequately protected from the radiological consequences of nuclear power reactor operation.

The United States Congress should undertake hearings and related actions necessary to:

1. **Resolve the PWR containment sump problem as soon as possible.**

   The PWR containment sump problem is well-known: NRC has been examining it since the 1970s.

   The PWR containment sump problem's solution is equally well-known: NRC already oversaw resolution of a very similar problem at the nation's boiling water reactors and the resolution of this specific problem at the PWR at the Davis-Besse nuclear plant in Ohio. Literally dozens of PWRs outside the United States have already resolved the problem.

   The NRC must cease its accommodation game with the nuclear industry and join the rest of the planet in fixing the PWR containment sump problem.

2. **Review close-outs of other generic safety issues to determine if there are any other improperly closures.**

   The PWR containment sump problem was initially handled by the NRC as Unresolved Safety Issue A-43, which they closed in 1985 without requiring any actions to be taken by plant owners. It is abundantly clear that NRC erred in its close-out process on this safety issue; hence the resurrection of this problem as GSI-191 in September 1996. All other safety issues that were closed without requiring any actions to be taken must be reviewed to determine if the PWR containment sump problem was the NRC's only such mistake.

3. **Reform the NRC so that it is consistently enforcing regulations.**

   Some years ago, the NRC realized that its inspection program focused on the adequacy of procedures developed for nuclear plants and slighted how well workers followed those procedures. The NRC revamped its inspection procedures to examine outcomes as well as processes. Likewise, the NRC could develop metrics on its own outcomes. For example, the recent Lessons Learned Task Force report prepared by the NRC identified numerous regulatory failures to ensure compliance with regulatory requirements at the Davis-Besse nuclear plant. Such efforts should be supplemented by formal evaluations during special inspections, augmented inspections, and incident investigations conducted by the NRC in response to plant events to determine if inconsistent enforcement contributed to the events' occurrence and/or significance. But the point is that when the NRC does not assess its own enforcement effectiveness, it can never identify – and more importantly, never correct – any shortfalls.

4. **Reform the NRC so that it is resolving safety issues expeditiously**.

   The NRC has been moving towards what it calls risk-informed regulation and has developed plant-specific risk assessment models. For each safety issue, the NRC can determine how many operating reactors it potentially effects and at least a bounding estimate for the safety significance of the problem. This information would help the NRC prioritize and manage the resolution of multiple safety issues. For example:

Emerging Issue	Reactors Affected (number)	Potential Safety Impact (per reactor)[##]	Overall Safety Impact (per industry)
A	68	8	544
B	103	2	206
C	9	40	360

Such an approach would help the NRC manage the resolution of safety issues on both a macro and micro level. On the macro level, the approach would demonstrate that Emerging Issue B, while affecting the entire fleet of reactors, has less overall safety impact than either Issue B or Issue C, which only affect a subset of the fleet. On the other hand, the approach would reveal that taking three times as long to resolve Issue B than Issue A has adverse safety consequences. Thus, the approach would better enable NRC to apply resources where they yield the greatest safety benefit.

On the micro level, the approach would highlight, for example, that the nine reactors affected by Issue C have a potentially significant impairment that should be considered by the NRC when making regulatory decisions about these reactors. Increased awareness of such potential impairments make it less likely that the NRC would approve a measure increasing the chances of challenged the potentially impaired system at these reactors.

Over its history, the NRC has separated resolution of emerging or generic safety issues from oversight of operating reactors. Integration of safety issue resolution into reactor oversight would expedite resolution. Plant owners would have great incentives either to demonstrate that their reactors are not affected by the issue or to fix the problem so that it doesn't continue to complicate other operational activities.

The NRC could undertake these reforms without Congressional involvement. But it could have done so any time during the past three decades yet failed to do so. The Congress must get involved now to make these overdue reforms happen and happen soon.

[##] These unitless, arbitrary values are provided to illustrate the concept.

APPENDIX A: 10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

(a)(1)(i) Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding[*] must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. ***Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident*****.** ***Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded.*** *Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. [emphasis added]*

(ii) Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.

(2) ***The Director of Nuclear Reactor Regulation may impose restrictions on reactor operation if it is found that the evaluations of ECCS cooling performance submitted are not consistent with paragraphs (a)(1) (i) and (ii) of this section.*** *[emphasis added]*

(3)(i) ***Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50F from the temperature calculated for the limiting transient using the last acceptable model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F.*** *[emphasis added]*

[*] Indian Point Units 2 and 3 are both pressurized light-water nuclear power reactors fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding.

(ii) ***For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or licensee shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually as specified in §50.4. If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with §50.46 requirements.*** *This schedule may be developed using an integrated scheduling system previously approved for the facility by the NRC. For those facilities not using an NRC approved integrated scheduling system, a schedule will be established by the NRC staff within 60 days of receipt of the proposed schedule. Any change or error correction that results in a calculated ECCS performance that does not conform to the criteria set forth in paragraph (b) of this section is a reportable event as described in §§50.55(e), 50.72 and 50.73. The affected applicant or licensee shall propose immediate steps to demonstrate compliance or bring plant design or operation into compliance with §50.46 requirements. [emphasis added]*

(b)(1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200º F.

(2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

(c) As used in this section: (1) Loss-of-coolant accidents (LOCA's) are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and

including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system.

(2) ***An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs <u>and all other information necessary for application of the calculational framework to a specific LOCA</u>, such as mathematical models used, <u>assumptions included in the programs, procedure for treating the program input and output information</u>, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure****. [emphasis added]*

(d) The requirements of this section are in addition to any other requirements applicable to ECCS set forth in this part. The criteria set forth in paragraph (b), with cooling performance calculated in accordance with an acceptable evaluation model, are in implementation of the general requirements with respect to ECCS cooling performance design set forth in this part, including in particular Criterion 35 of appendix A. [30]

Sources:

[1] Atomic Safety and Licensing Appeal Board, Atomic Energy Commission, Memorandum and Order (ALAB-138), Section IV, paragraph A.1, July 31, 1973.
[2] United States General Accounting Office, GAO/RCED-87-141, "Nuclear Regulation: Efforts to Ensure Nuclear Power Plant Safety Can Be Strengthened," page 10, August 1987.
[3] Office of the Inspector General, Nuclear Regulatory Commission, Case No. 97-03S, "Public's Concerns with NRC Report of Independent Safety Assessment (ISA) at Maine Yankee," January 26, 1998.
[4] Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Director's Status Report on Generic Activities," April 2003.
[5] Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, "Director's Status Report on Generic Activities," April 2003.
[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," Table 5-7, "Results of Parametric Evaluations Regarding Potential for Blockage," August 2001.
[7] Nuclear Regulatory Commission, Generic Letter 85-22, "Potential For Loss Of Post-LOCA Recirculation Capability Due To Insulation Debris Blockage," December 3, 1985.
[8] Nuclear Regulatory Commission, Generic Letter 85-22, "Potential For Loss Of Post-LOCA Recirculation Capability Due To Insulation Debris Blockage," December 3, 1985.
[9] Nuclear Regulatory Commission letter from John A. Grobe, Chairman – Davis-Besse Oversight Panel, to Lew W. Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station NRC Integrated Inspection Report 50-346/03-17," page 39 of the enclosure, September 29, 2003.
[10] NRC Letter dated October 7, 2003, from James L. Caldwell, Regional Administrator, to Lew Myers, Chief Operating Officer, FirstEnergy Nuclear Operating Company, "Final Significance Determination for a Yellow Finding (NRC Inspection Report 50-346/03-15) – Davis-Besse Potential Clogging of the Emergency Sump Following a Loss of Coolant Accident."
[11] Atomic Safety and Licensing Appeal Board, Atomic Energy Commission, Memorandum and Order (ALAB-138), Section IV, paragraph A.1, July 31, 1973.
[12] Nuclear Regulatory Commission, Information Notice 96-17, "Reactor Operation Inconsistent with the Updated Final Safety Analysis Report," March 18, 1996.
[13] Union of Concerned Scientists, "Safety Second: A Critical Evaluation of the NRC's First Decade," February 1985.
[14] United States General Accounting Office, GAO/RCED-87-141, "Nuclear Regulation: Efforts to Ensure Nuclear Power Plant Safety Can Be Strengthened," pp. 34-35, August 1987.
[15] Letter from Alex Matthiessen, Executive Director – Riverkeeper, Inc., and David Lochbaum, Nuclear Safety Engineer – Union of Concerned Scientists, to William D. Travers, Executive Director for Operations, Nuclear Regulatory Commission, "Indian Point Energy Center – Petition Pursuant to 10 CFR 2.206 – PWR Containment Sump Failure," September 8, 2003.
[16] Nuclear Regulatory Commission, EA-03-086, "Issuance of Order Requiring Compliance with Revised Design Basis Threat for Operating Power Reactors," April 29, 2003.
[17] Nuclear Regulatory Commission from Chairman Joseph Hendrie to President Jimmy Carter, "Annual Report, 1978," page 38, February 14, 1979.
[18] Nuclear Regulatory Commission memo from A. W. Serkiz, Task Manager, to Distribution, "Request for Review of NUREG-0869, Revision 1," February 3, 1984.
[19] Nuclear Regulatory Commission memo from A. W. Serkiz, Task Manager, to Distribution, "Request for Review of NUREG-0869, Revision 1," February 3, 1984.
[20] Nuclear Regulatory Commission memo from A. W. Serkiz, Task Manager, to Distribution, "Request for Review of NUREG-0869, Revision 1," February 3, 1984.
[21] Nuclear Regulatory Commission, Generic Letter 85-22, "Potential For Loss Of Post-LOCA Recirculation Capability Due To Insulation Debris Blockage," December 3, 1985.
[22] Nuclear Regulatory Commission, Generic Letter 85-22, "Potential For Loss Of Post-LOCA Recirculation Capability Due To Insulation Debris Blockage," December 3, 1985.
[23] Nuclear Regulatory Commission, Transcript, "Discussion of Proposed Rule on Backfitting," page 29 lines 15-25, May 22, 1984.

[24] Ivan Selin, Chairman, Nuclear Regulatory Commission, Speech before the Regulatory Information Conference, "The Future of Reactor Regulation," pp 7-8, May 9, 1995.
[25] Nuclear Regulatory Commission letter from Harold R. Denton, Director – Office of Nuclear Reactor Regulation, to Chairman and Commissioners, "Compliance of NRC Licensees with NRC Regulations, Regulatory Guides, Branch Technical Positions, and Licensee Commitments," July 23, 1980.
[26] Subcommittee on General Oversight and Investigations of the Committee on Interior and Insular Affairs, United States House of Representatives, "Relationship of the Nuclear Regulatory Commission to the Nuclear Industry," written testimony of Loren L. Bush, Jr., Chief – Program Development and Review Section, Safeguards Branch, Division of Reactor Inspection and Safeguards, Nuclear Regulatory Commission, June 11, 1987.
[27] John G. Kemeny, "Saving American Democracy: The Lessons of Three Mile Island," *Technology Review*, June/July 1980, page 65.
[28] *Federal Register*, Vol. 50, No. 183, "Separate Views of Commissioner Asselstine," September 20, 1985, page 38106
[29] Nuclear Energy Agency Committee on the Safety of Nuclear Installations, NEA/CNSI/R(2002)6, "Final Report: Knowledge Base for Strainer Clogging – Modifications performed in different countries since 1992," October 3, 2002.
[30] 10 CFR 50.46. [39 FR 1002, Jan. 4, 1974, as amended at 53 FR 36004, Sept. 16, 1988; 57 FR 39358, Aug. 31, 1992; 61 FR 39299, July 29, 1996; 62 FR 59726, Nov. 3, 1997]