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DEVONRUE

38 North Street
Suite 200
Hingham, Massachusetts 02043
Voice: 781-740-1445
FAX: 781-740-1465
Website: www.devonrue.com

May 14, 1998
DLNRC 9805-01/SM

Mr. Timothy E. Collins
Office of Nuclear Reactor Regulations
NRR/DSSA/SRXB, Mail Stop OWFN/8E23
United States Nuclear Regulatory Commission
Washington, District of Columbia 20555

Subject: Indications Plants Licensed Using Siemens LOCA Methodology May Be Operating in Violation of 10 CFR 50.46

Dear Mr. Collins:

Mr. Eugene Imbro, NRC Deputy Director for ICAVP, suggested you might have certain technical information related to the May 12, 1998 public meeting concerning Millstone Unit 2. Not knowing the proper protocols for pursuing this information, I present this letter to you and ask you route it to the appropriate individual if you are not that person.

Among other things, emergency core cooling systems (ECCS) are required to perform in such a manner that the maximum or peak fuel element cladding temperature (PCT) does not exceed 2200F. This limit is intended to prevent fuel failure in a reactor accident. Changes or errors affecting PCT calculations are reportable when discovered in an evaluation model found acceptable to NRC. If the change or error is significant, the licensee report must be filed within 30 days, and include a proposed schedule for reanalysis or other action to comply with NRC safety requirements in a reactor accident.[1]

At the May 12, 1998 public meeting concerning Millstone Unit 2 ICAVP, Northeast Utilities (NU) engineers reported the Siemens methodology is expected to change due to errors of an unspecified nature.[2] It was further reported, apparently for some number of years, these errors masked the correct PCT for a reactor accident at plants licensed with Siemens' methodology -- PCT in fact exceeds NRC's 2200F safety limit. Notably, the higher PCT revelation is not unique to Millstone Unit 2; it applies to <u>all</u> plants licensed with the Siemens methodology.[3]

Since PCT values above 2200F violate Federal safety regulations, and could seriously exacerbate the public health consequences of a reactor accident, this information is significant and demands immediate action to protect the public health and safety.

NU also reported that Siemens filed a "Part 21," though the report was not specifically identified.[4] In fact, Siemens filed a Part 21 report on February 23, 1998 alerting the agency to mathematical errors in the RELAP4 model of large-break loss of coolant accidents (LBLOCA).[5] NU appeared to suggest Siemens might have been aware since at least March 1998 that these errors affect a sizeable portion of the US reactor fleet.[6] Evidently under the Millstone Unit 2 ICAVP effort, Siemens has been working with NU staff to identify and correct errors in its reactor accident analysis methodology. Were it not for NU's ICAVP activities, it is conceivable Siemens' errors would never have surfaced.

1 "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors," 10 CFR 40.46
2 See Page 5, Northeast Utilities handout, "Reanalysis Scope," stating: "A Siemens methodology change is likely ..."
3 See Page 11, Northeast Utilities handout, "Reanalysis Scope," stating: "Generic issue associated with variability issues in the Siemens LBLOCA methodology."
4 "Notification of failure to comply or existence of a defect and its evaluation," 10 CFR 21.21
5 See NRC Headquarters Daily Report, February 23, 1998
6 See Page 11, Northeast Utilities handout, "Reanalysis Scope," stating: "Reported under Part 21."

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It is commendable that Siemens has been both willing and able to enter the industry's "Holy of Hollies" – PCT predictions in a reactor accident – and flush out into the open longstanding errors. Apparently, these errors are not restricted to either PCT nor are they recent.

For example, Siemens filed a Part 21 about a year ago reporting errors in accident modeling affecting "certain plants," which are not identified.[7] More recently, Siemens alerted the agency to significant nonconservatisms affecting two US boiling water reactors.[8] With such a pattern of reports, there is often a tendency to shoot the messenger. Still, Siemens' willingness to actually follow the thread – and with effect – speaks far louder than the industry's apparent service mark of "commitment to excellence."

At the same time, it is chilling to think so many years elapsed with the PCT landmines in place. Apparently, serious technical housecleaning only happens when a new vendor, foreign-owned at that, ventures upon the scene, assumes "verbatim compliance" means just that, and kicks (trips?) over a rock. Millstone Unit 2's validation experience also suggests what might actually be the rule in this industry, rather than the exception. All told, not an especially inspiring image, by any stretch.

NRC scheduled a public meeting with Siemens for May 21, 1998 at which time additional information may become available. However, it was evident at the NU meeting Siemens will not correct errors in its PCT calculations and other studies until mid-late June – 3 months or more since Siemens first plumbed their depths. This schedule does not include NU's review and validation efforts. Thus, it seems another 6-8 weeks must elapse before necessary changes to the Millstone Unit 2 operating license can even be identified. This period does not include NU's proposed modifications requiring NRC action, time needed to determine the radiological consequences to the public, or time provided the public for comment about all of this. Who knows what, if anything, is going on at the operating reactors.

For numerous reasons, but including these PCT errors, Millstone Unit 2 is shutdown. Errors in its operating license are of no immediate consequence to the public safety. That may be all the good news there is.

Unfortunately, plants licensed to the Siemens methodology are currently operating without restriction. When will they get around to comparable state of knowledge? Given the safety significance of the PCT error, it is truly dichotomous that 10 CFR 50.46 only applies to a shutdown reactor – one that is apparently the sacrificial lamb – Millstone Unit 2.

For such situations, NRC's requirements for operating plants are a simple matter of law, and should leave no room for enforcement discretion:

> " ... propose **immediate steps** to demonstrate compliance or bring plant design or operation into compliance with §50.46 requirements."[9] (emphasis added)

The remedies for licensees failing to comply with safety requirements are equally clear, as at least Millstone Unit 2 knows so well:

> "**The Director of Nuclear Reactor Regulations 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."[10] (emphasis added)

As they say, this seems like a "no-brainer." NRC need not look very far for guidance on what must be done.

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7 "Critical Heat Flux Database for 9x9 Fuel Designs Does Not Adequately Estimate Uncertainties," NRC Event 32379, May 22, 1997

8 "Nonconservative Predictions of Monitored MCPR," NRC Event 34119. April 21, 1998

9 10 CFR 50.46(a)(3)(i)

10 10 CFR 50.46(a)(2)

So, what has been done? I confess that I don't have a clue, despite the fact it's not every day such a long and broad shadow is suddenly thrown across LOCA analyses.

The standard seems clear enough. Agencies regularly take prompt action to address uncertain circumstances potentially affecting the public safety. For example, the potential for frayed insulation on electrical wires in Boeing 737s led to the removal of a major portion of that fleet from revenue service until such time inspections resolved their safety status. If repairs are needed, the aircraft remains out of revenue service until completed. Such prompt action is entirely appropriate given the potential loss of life that could result from an accident triggered by an electrical short.

As they say, a "no-brainer."

In the case of PCT temperatures, we may be dealing with a far more serious situation than the potential for frayed wire leading to an electrical short on commercial aircraft and igniting abnormally high fuel vapors.

With rising temperatures in a LOCA, well-established chemical and metallurgical processes cause fuel rods to swell and eventually burst, spilling fission products into the coolant, out through the pipe break, and beyond. The cladding alloy reacts with steam at accelerating rates as accident temperatures rise. Above 2200F, a positive feedback loop develops -- a "snowballing effect" -- in which small changes in nuclear fuel rod temperatures result in large changes in fuel damage and still higher fuel rod temperatures. The accident at TMI-2 and fuel damage research demonstrates the time above such temperature limits need not be very long before the deadly feedback loop is triggered and serious core damage results.

We are not discussing the ethereal such as the safety significance of a "leading" performance indicator, or some PRA scenario pencil-whipped down to the 1×10^{-5} RY^{-1} threshold. LOCA protection is a matter of law. Licenses will no doubt be amended and plants modified to reduce the PCT. This is real nuclear safety.

There is also important precedent in the policies that evolve. If Federal LOCA requirements can be so casually violated without swift, very public, and the most serious agency action, why should NRC enforce <u>any</u> of the current portfolio of safety requirements? If FAA can quickly pull this nation's 737 fleet out of revenue service to inspect wires <u>suspected</u> of damage, NRC's course of action should be no less clear for <u>known</u> defects.

In speaking with Mr. Egan Wang yesterday, NRC's contact for the upcoming Siemens meeting, it was obvious neither of us possess a complete set of essential technical information concerning this potentially serious policy and safety issue. Therefore, I request the following information at your earliest convenience, and as each piece becomes available:

(1) A listing of all nuclear facilities with a license amendment relying upon the Siemens' methodology.

(2) A copy of the Siemens' Part 21 report and other reports and correspondence filed by Siemens and licensees concerning the PCT errors and issues associated with ECCS performance.

(3) Justification provided by NRC, Siemens, or the affected licensees for continued power operations while ECCS is inoperable for LOCA and other design basis accidents.

(4) An explanation how the errors identified by Siemens could have remained undiscovered in US reactors all these years.

(5) An assessment of the likelihood that comparable errors exist among other safety analyses, including those of other vendors.

(6) What NRC intends to do to ensure the current Siemens PCT problem does not recur.

(7) The actions under 10 CFR 50.46 your staff recommends, or your schedule for providing such recommendations, to the Director of Nuclear Reactor Regulations.

You might also consider posting this information in a special place on NRC's website.

I appreciate your consideration of this matter. I realize even considering my questions diverts you from pressing and important safety duties, and I apologize for any inconvenience.

As a matter of courtesy, I am providing a copy of this letter to Mr. Eugene Imbro, NRC's senior representative at the May 12, 1998 NU meeting. Mr. Imbro was very gracious in answering my questions, despite his rather tight travel plans that day.

Given Dr. Jackson's special interest in NU's design basis validation efforts and their lessons for nuclear safety, I am also forwarding a copy to her office.

Sincerely,

STEPHEN MALONEY

Enclosure Northeast Utilities Handout, "Reanalysis Scope," presented at Millstone Unit 2 ICAVP Public Meeting, Northeast Utilities Corporate Offices, Berlin Connecticut, May 12, 1998

Cc: Dr. Shirley Jackson, Chairman, US Nuclear Regulatory Commission, w/enclosure
Mr. Eugene V. Imbro, US Nuclear Regulatory Commission, w/enclosure

Via: Facsimile
US Express Mail

REANALYSIS SCOPE

Issue	Analysis Affected
A. MSSV and piping pressure drop	1. Loss of Load 2. Inadvertent MSIV Closure
B. Auxiliary Feedwater Flow	<u>Maximum Flow</u> 1. Containment Analysis (Steam Line Break (SLB)) 2. SLB return to power <u>Minimum Flow</u> 1. Loss of Feedwater - Chapter 14 2. Loss of Feedwater - Chapter 10 3. Small Break LOCA 4. Steam Generator Tube Rupture (SGTR)

REANALYSIS SCOPE

Issue	Analysis Affected
C. HPSI flow	1. SLB return to power 2. Small Break LOCA 3. Large Break LOCA 4. SGTR
D. LPSI flow	1. LBLOCA
E. Charging flow	1. Boron Dilution 2. SGTR
F. RPS uncertainty	1. Uncontrolled Rod Withdrawal from a low power or subcritical condition 2. SLB inside containment 3. Increased Steam Flow

REANALYSIS SCOPE

Issue	Analysis Affected
G. Containment heat removal systems (CAR fans, containment spray, RBCCW)	1. Containment Analysis (SLB and LBLOCA) 2. Large Break LOCA 3. SLB inside containment

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
1. Increase in Steam Flow	Analysis in Progress	06/15/98	No	1. Analysis will be included in SLB report. 2. Addresses NI decalibration due to temperature effects.

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
2. SLB outside containment	Analysis in progress	06/15/98	Yes	1. A number of NCRs have been generated by Siemens. 2. A Siemens methodology change is likely; NU will make a submittal. 3. Preliminary results indicate fuel failure at return to power.

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
3. SLB inside containment	Analysis in progress	06/15/98	No	1. Preliminary results show that the offsite power available cases is bounded by outside containment results. 2. Preliminary results indicate that SLB coincident with loss of offsite power show failed fuel due to harsh environment uncertainty applied to low flow trip. 3. Radiological Consequences bounded by Control Rod Ejection Accident

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
4. Loss of Load	Complete	----------	No	FSAR Change in Progress
5. Inadvertent MSIV Closure	Complete	---------	No	FSAR Change in Progress
6. Loss of Feedwater - Chapter 14	Analysis in Progress	05/29/98	No	Analysis results acceptable but show little margin to SG dry out due to reduction in AFW flow and reduced SG inventory with the replacement steam generators.

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
7. Loss of Feedwater - Chapter 10	Analysis in Progress	----------	No	1. Best estimate analysis needed to support AFW reliability and availability requirements and is the basis for manual turbine driven AFW pump start. 2. Assumptions are still being developed to achieve success.

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
8. Uncontrolled Rod Withdrawal from Subcritical	Analysis in Progress	05/25/98	No	Analysis redone to address longer RPS response time and CPC uncertainty increase at low power.
9. Boron Dilution	Analysis in Progress	05/22/98	No	1. Analysis redone for a higher dilution flow rate based upon maximum charging capacity. 2. Revised SDC flow uncertainty

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
10. Small Break LOCA	Analysis in Progress	06/15/98	No	1. Preliminary results show increased sensitivity to loop seal clearing because of the reduction in HPSI and AFW flow. 2. Credit is needed for charging.

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
11. Large Break LOCA	Analysis in Progress	--------------	Yes (Independent of Appendix K analysis)	1. Generic issue associated with variability issues in the Siemens LBLOCA methodology. 2. Reduction in Linear Heat Generation rate necessary. 3. Reported under Part 21 4. Siemens/ NRC meeting planned for 05/21/98

Status of Reanalysis

Analysis	Status	Best Estimate Schedule	Radiological Analysis Required	Comments
12. Containment Analysis	Analysis Complete	-------------		FSAR change in progress
13. SGTR	Analysis in Progress	05/29/98	Yes	Analysis being updated to reflect changes in charging AFW and HPSI

MILLSTONE UNIT 2

REANALYSIS SCOPE - DOSE ANALYSIS

ACCIDENT	IN FSAR NOW
1. LOSS OF COOLANT ACCIDENT	
a. OFF SITE	YES
b. MP2 CONTROL ROOM	YES
c. MP3 LOCA TO MP2 CONTROL ROOM	YES
2. MAIN STEAM LINE BREAK	NO
3. STEAM GENERATOR TUBE RUPTURE	YES
4. ROD EJECTION ACCIDENT	NO
5. FUEL HANDLING ACCIDENT	YES
6. CASK DROP	YES

Status for the above reanalysis: Input parameters are being finalized to allow Stone and Webster to begin analysis.

Best estimate schedule: July 31, 1998