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July 5, 2000

Office of the Secretary
ATTN: Rulemaking and Adjudicatory Staff
U.S. Nuclear Regulatory Commission
Washington DC 20555-0001

Re: **In the Matter of Northeast Nuclear Energy Company (Millstone Nuclear Power Station, Unit No. 3)**
Docket No. 50-423-LA-3/ASLBP No. 00-771-01-LA

Dear Sir:

Enclosed please find the original and two (2) copies of the corrected "Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone Intend to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Liensee Regarding the Proposed Expansion of Spent Fuel Storage at the Millstone Unit No. 3 Nuclear Power Plant."

Please note that the corrected submission reorders Exhibit Nos. 7 et seq.

Thank you for your assistance.

Sincerely,

Nancy Burton

Nancy Burton, Esq.

cc: Service List w/att.

Template = SECY-021

SECY-02

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	:	**Docket No. 50-423-LA-3**
	:	
	:	**ASLBP No. 00-771-01-LA**
	:	
Northeast Nuclear Energy Company	:	
	:	
(Millstone Nuclear Power Station,	:	
Unit No. 3)	:	**July 3, 2000**

DETAILED SUMMARY OF FACTS, DATA AND ARGUMENTS AND SWORN SUBMISSION ON WHICH CONNECTICUT COALITION AGAINST MILLSTONE AND LONG ISLAND COALITION AGAINST MILLSTONE INTEND TO RELY AT ORAL ARGUMENT TO DEMONSTRATE THE EXISTENCE OF A GENUINE AND SUBSTANTIAL DISPUTE OF FACT WITH THE LICENSEE REGARDING THE PROPOSED EXPANSION OF SPENT FUEL STORAGE CAPACITY AT THE MILLSTONE UNIT NO. 3 NUCLEAR POWER PLANT

I. INTRODUCTION

Connecticut Coalition Against Millstone ("CCAM") and Long Island Coalition Against Millstone ("CAM") (collectively, the "Intervenors") hereby submit a detailed written summary and sworn submission (hereinafter "Summary") of all the facts, data and arguments which are known to the Intervenors and on which the Intervenors propose to rely at the July 19, 2000 oral argument, pursuant to 10 C.F.R. Sec. 2.1113.

This Summary presents the Intervenors' factual and legal grounds for asserting that Northeast Nuclear Energy Company's ("NNECO's") application to amend the Millstone Unit 3 Operating License by expanding the capacity of its spent fuel storage pool poses an undue and unnecessary risk of a criticality accident, increases the probability of a criticality accident, and fails to satisfy the criticality prevention requirements of General Design Criterion ("GDC") 62 because it improperly relies on administrative controls.

As required by 10 C.F.R. Sec. 2.1113, the factual assertions in this Summary are submitted under the sworn declaration of Dr. Gordon Thompson, the Intervenors' expert witness in this proceeding.

As detailed below, this Summary demonstrates that NNECO's License Amendment Application must be rejected because it places impermissible reliance on administrative procedures and controls for criticality prevention, rather than relying entirely on physical systems and processes, as required by the regulations. The Intervenors assert that they have submitted substantial evidence to support the existence of a genuine and substantial dispute of fact, pursuant to 10 C.F.R. Section 2.1113.

II. STATEMENT OF THE CASE

This case challenges the licensee's proposal to trade reliance on physical protection for administrative controls to an extent that poses an undue and unnecessary risk of criticality accident. This case also raises questions about the proper interpretation of GDC 62, which requires that criticality in the fuel storage and handling system of a nuclear power plant must be prevented by "physical systems and processes, preferably by use of geometrically safe configurations." This regulation clearly precludes the use of such administrative controls and procedures as control of burnup/enrichment levels and reliance on the presence of soluble boron in fuel pools. Although the NRC Staff's current regulatory guidance countenances the use of such administrative controls, it must be disregarded in this respect because it is fundamentally inconsistent with the controlling regulation, GDC 62.

The trade-off of physical protection for administrative controls has particular resonance with regard to the licensee in question, Northeast Nuclear Energy Company, which has been associated with failure of application and enforcement of administrative controls. Yet, the proposed application seeks to impose a greater burden on the workforce to carry out more complex tasks simply to achieve a higher density of storage of waste onsite at Millstone Unit 3.[1] This Summary will demonstrate that the new administrative controls do involve greater complexity and, hence, greater potential for human error; that failure of administrative controls can lead to a criticality accident; that criticality calculations can contain errors; that fuel can be mispositioned; and that dilution of soluble boron can occur. Moreover, this Summary will demonstrate that criticality calculations at Millstone have contained errors; that fuel has been mispositioned at Millstone; and circumstances exist at Millstone whereby dilution of soluble boron could occur.[2] This Summary will demonstrate that the technical analysis submitted in support of the license amendment is insufficient and that neither the licensee nor the NRC Staff has a proper understanding of the increased burden of risk involved.

The Intervenors submit that the License Amendment Application poses an undue and unnecessary risk of a criticality accident. They further submit that the NRC lacks legal authority to approve the application because it fails to comply with GDC 62. Finally, the Intervenors submit that neither NNECO nor the NRC Staff has demonstrated that the public health and safety will be adequately protected if the amendment is granted.
The Intervenors submit that the License Amendment Application poses an undue and unnecessary risk of a criticality accident. They further submit that the NRC lacks legal authority to approve the application because it fails to comply with GDC 62. Finally, the Intervenors submit that neither NNECO nor the NRC Staff has demonstrated that the public health and safety will be adequately protected if the amendment is granted.

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[1] Condition Report #M3-99-1148 suggests that the expansion is sought in part to provide additional storage capacity for Unit 2 spent fuel as well. Exhibit 1.

[2] This Summary incorporates by reference all discovery materials produced by NNECO and the NRC Staff, as well as the Intervenors. Certain of such exhibits are appended to this Summary.

III. FACTUAL AND PROCEDURAL BACKGROUND

A. History of Criticality Prevention at Nuclear Power Plants

1. Nature of Criticality Accidents

In operating a nuclear power plant, it is necessary to protect the facility against a criticality accident. Criticality occurs when neutrons emanating from atoms of special nuclear material, as a result of fission of their nucleii, bombard other atoms and cause fission of their nucleii, setting off a chain reaction. Criticality can be prevented by providing adequate spacing of special nuclear material, and by introducing neutron-absorbing material to shield the special nuclear material and absorb the neutrons.

A nuclear fission reactor generates power because criticality is achieved under controlled conditions. At all times when fresh or spent fuel is outside a reactor, criticality must be prevented. In the case of light-water reactor fuel, a criticality event can occur if fresh or spent fuel assemblies are brought sufficiently close together in the presence of a neutron-moderating material such as water, without the presence of sufficient neutron-absorbing material to suppress criticality. The neutron-absorbing material could be solid boron or other material incorporated into the structure of the racks where fuel assemblies are stored, or soluble boron in the water surrounding fuel assemblies.

2. Regulations and agency guidance

Criticality control at nuclear power plants is governed by General Design Criterion ("GDC") 62, which requires that:

> Criticality in the fuel handling and storage system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

10 C.F.R. Part 50, Appendix A, Criterion 62. This language clearly precludes the use of ongoing procedural or administrative controls for criticality prevention.[3] The NRC also has regulations at 10 C.F.R. § 70.24 and § 50.68, which permit licensees to forego criticality monitors if they comply with certain measures for criticality prevention. As discussed in more detail in Section VI.D. below, these measures are consistent with GDC 62, and the Commission reaffirmed GDC 62 when it promulgated the regulations.

GDC 62 sets forth unequivocal requirements for the prevention of criticality under normal conditions. However, one can postulate accident conditions that would defeat these requirements. For example, a sufficiently severe mechanical loading could reduce the center-to-center distance between fuel assemblies and thereby cause criticality, even though the configuration was geometrically safe before the loading was applied.

In 1978, the NRC Staff issued a guidance document which sought to extend the requirements of GDC 62 into the realm of accident conditions, by introducing the "Double Contingency Principle" and the concept of "realistic initial conditions."[4] The guidance is attached to a letter from Brian K. Grimes of the NRC Staff to "All Power Reactor Licensees," dated April 14, 1978 (hereinafter "Grimes Letter").[5] The Grimes letter acknowledges that "[d]ue to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better

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3 For a more complete discussion of the language and history of GDC 62, see Section VI. below.

4 *See* Appendix A to this Summary for a further discussion of the source and development of these terms.

5 A copy of the Grimes Letter is attached as Exhibit 2.

utilize available storage space."[6] The Letter provides the following guidance for evaluation of criticality prevention under postulated accident conditions:

> The double contingency principle of ANSI N 16.1-19754 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.
>
> Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.[7]

As discussed in Appendix A, these terms are not further discussed or defined in the Grimes Letter. However, it is clear that the Grimes Letter did not allow reliance on the presence of soluble boron as a criticality prevention measure under normal conditions. Instead, the presence of soluble boron was intended to be considered solely as an initial condition in an accident scenario.

In 1981, the Staff issued a draft regulatory guide containing further guidance for the evaluation of criticality prevention measures: Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis (December 1981) (hereinafter "Draft Reg. Guide 1.13")[8]. Although Draft Reg. Guide 1.13 has never been issued in final form, the Staff has applied it extensively to the review of spent fuel pool expansion applications. Like the 1978 Grimes Letter, this Draft Reg. Guide has never been approved by the Commission, but is solely a Staff guidance document.

In §§ 4.5 and 6 of Appendix A, Draft Reg. Guide 1.13 implies that credit may be taken for fuel burnup as a criticality prevention measure under normal conditions. Section 5.2 of Appendix A states that the presence of soluble boron can be regarded as a realistic initial condition under certain accident conditions, namely those associated with

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6 *Id.*, Enclosure 1 at I-1.

7 *Id.*

8 A copy of Draft Reg. Guide 1.13 is attached as Exhibit 3.

"Condition IV faults," which are not defined in the Draft Reg. Guide. As in the case of the Grimes Letter, it is clear that this Draft Reg. Guide does not allow reliance on the presence of soluble boron as a criticality prevention measure under normal conditions.[9] Draft Reg. Guide 1.13 also calls for the application of the Double Contingency Principle, articulating the principle as follows:

> At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent and concurrent failures or operating limit violations.

Appendix A, § 1.4 (emphasis in original). The Draft Reg. Guide's version of the Double Contingency Principle is broadly consistent with the language of the Grimes Letter, although there are two notable differences, the first of which strengthens the standard significantly. First, § 1.4 specifies "at least two" criticality-inducing events, whereas the Grimes letter specifies "two" events. Second, § 1.4 refers to "failures or operating limit violations," while the Grimes Letter refers to "events."

A more recent guidance document on criticality prevention in spent fuel storage pools is a Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants (August 19, 1998) (hereinafter "Kopp Memorandum").[10] The Kopp Memorandum asserts the Staff's acceptance of various administrative measures for criticality prevention, such as credit for burnup and soluble boron. It also re-states, in substantially weakened form, the Double Contingency Principle:

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9 As discussed in Attachment A to this Summary, the American Nuclear Society ("ANS") also provides guidance regarding the presence of soluble boron as an initial condition for the purposes of criticality analysis pertinent to accident conditions.

10 A copy of the Kopp Memorandum is attached as Exhibit 4.

> The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.[11]

The Kopp Memorandum thus effectively reduces the double contingency principle to a "single contingency principle."[12]

Thus, as the pressure has increased for higher and higher density fuel storage, the NRC Staff has increasingly relaxed the standards for criticality prevention, allowing the use of administrative controls and reducing the rigor of the accident analysis required.

3. Evolution of Criticality Prevention in Fuel Pools

There is no centralized, publicly accessible database that provides detailed information about the rack configuration at each nuclear power plant spent fuel storage pool and the history of rack installation at each pool. Nevertheless, a survey of correspondence and safety reports for individual plants shows how measures for criticality prevention at nuclear power plants have evolved over time in response to increasing demand for higher and higher density spent fuel storage. This evolution has gone beyond the bounds of measures that are consistent with GDC 62. The NRC Staff has condoned violations of GDC 62 by issuing regulatory guidance that countenances these violations, and by approving many license amendment applications that permit the use of administrative controls for criticality prevention in the high-density storage of spent fuel.

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11 *Id.*, Attachment 4.

12 A more detailed discussion of the Kopp Memorandum appears in Appendix A to this Summary.

a. Low-density storage

When U.S. nuclear power plants of the present generation were designed, and when many of the currently operating plants were commissioned, fuel pools were equipped with low-density fuel storage racks. The racks were designed with open frames of steel or aluminum. Center-center distances between fuel assemblies were typically 10-13 inches in BWR racks and 18-22 inches in PWR racks. By using a relatively low fuel storage density -- less than 0.25 tonne U per square foot -- licensees achieved a high level of safety against criticality. The center-center distances were large enough to prevent criticality even if fresh fuel was placed in the racks and the pool was filled with unborated water. In other words, criticality prevention relied entirely on the use of a geometrically safe configuration.

As spent fuel began to accumulate at power plants, there was growing interest in achieving higher storage densities in fuel pools. This implied smaller center-center distances in the racks, resulting in a greater propensity for criticality. Beginning in the 1970s and continuing through the 1980s and 1990s, center-center distances in fuel pools were reduced in several steps. Additional means of criticality prevention were introduced at each step.[13]

b. Reliance on the neutron-absorbing properties of storage racks and the incorporation of flux traps

The first step toward higher density was to employ stainless steel racks with center-center distances of about 8 inches in BWR racks and 13 inches in PWR racks. Roughly speaking, this step occurred in the 1970s. The new configuration increased the fuel storage density to a level of up to 0.39 tonne U per square foot. The reduced center-center distances in this configuration yielded a greater propensity for criticality than was exhibited by the previous open-frame racks. Nevertheless, the rack designers were able to achieve a subcritical margin of reactivity, relying in part on the absorption of slow neutrons by the stainless steel in the rack structures. This neutron-absorption phenomenon was in turn assisted by the moderation of fast neutrons by water confined in passages ("flux traps") between the fuel assemblies. At this stage of evolution in fuel storage density, criticality prevention relied partly on the distance between fuel assemblies and partly on the neutron-absorbing properties of the racks.

c. Incorporation of boron in the structure of storage racks

The second step toward higher density in fuel pools was to employ stainless steel racks which incorporated boron in solid form within the rack structures. Roughly speaking, this step occurred in the 1980s. Boron is an absorber of neutrons, and thereby suppresses criticality. Thus, the incorporation of solid boron allowed center-center distances to be further reduced. A common method of incorporating solid boron is to attach Boral panels to the racks. To construct a Boral panel, boron carbide is dispersed in aluminum, and this material is fabricated into sheets which are clad with aluminum. These "panels" are then attached to the spent fuel storage racks.

Incorporation of solid boron within the rack structures allowed a subcritical margin of reactivity to be maintained while center-center distances were reduced to 6.5 inches in BWR racks and 10.5 inches in PWR racks, thereby achieving a fuel storage density up to 0.58 tonne U per square foot. In this configuration, criticality prevention relied to a lesser degree than previously on the distance between fuel assemblies and to a greater degree on the neutron-absorbing properties of the racks. [14] Most, perhaps all,

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[13] *See* U.S. Department of Energy, Spent Fuel Storage Fact Book, DOE/NE-0005, April 1980; and USNRC, Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel, NUREG-0404 (2 volumes) Appendices B and D (March 1978).

[14] In pursuit of even higher storage densities in fuel pools, the nuclear industry has also studied fuel storage options involving a reduced presence of water between the fuel rods. Water moderates fast neutrons, so a reduced presence of water can yield a subcritical margin of reactivity even as the spacing between fuel assemblies or rods is reduced. One water-displacing option is to place spent fuel assemblies inside cans and to fill all empty space inside each can with small metal beads, thereby achieving a fuel storage density of 0.75 tonne U per square foot. A second option is to compact fuel assemblies by crushing the fuel spacers until rods are nearly touching, thus achieving a fuel storage density of about 0.95 tonne U per square foot. A third option is to dismantle the fuel assemblies and store the rods in close contact with each other inside cans, thus achieving a fuel storage density of about 1.1 tonne U per square foot. None of these

fuel pools at US nuclear plants have been equipped for some years with racks that incorporate solid boron within the rack structures, often in the form of Boral panels.

d. Ongoing administrative controls

In recent years, a number of licensees have further increased the density of spent fuel pool rack storage. As the fuel is packed closer and closer together, fixed neutron-absorbing material such as Boral panels becomes less and less effective in preventing criticality. Therefore, licensees have introduced ongoing administrative procedures for criticality prevention. These controls consist of (a) relying on the presence of soluble boron into the spent fuel pool water, (b) controlling the burnup level of the fuel, and (c) controlling the age of the fuel assemblies. Using these ongoing administrative controls, the density of storage of intact fuel assemblies in fuel pools has been increased beyond the level that was achieved by adopting center-center distances of 6.5 inches in BWR racks and 10.5 inches in PWR racks.

These three methods exploit phenomena as follows. First, increased burnup of a fuel assembly will, over a broad range of conditions, decrease the assembly's reactivity because of the ingrowth of neutron-absorbing isotopes and the reduced enrichment in U-235 that occur with increased burnup.[15] Second, the presence of soluble boron in the pool water will decrease reactivity because the soluble boron absorbs neutrons. Third, aging of a fuel assembly will decrease the assembly's reactivity due to the decay of Pu-241 (with a 14-year half-life) and the ingrowth of its decay product Am-241.

e. Independent Spent Fuel Storage Installations

There is an alternative to adopting ever-higher densities of fuel storage in an existing fuel pool. That alternative is to construct an independent spent fuel storage installation ("ISFSI"). ISFSI's have been built at several US nuclear plant sites. In each

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options has been generally adopted. *See* U.S. Department of Energy, Spent Fuel Storage Fact Book, DOE/NE-0005 (April 1980).

15 Burnup is the accumulated fission energy released by a fuel assembly. Its effects on criticality are exploited by restricting the combined burnup/enrichment parameters of fuel assemblies that are placed in the fuel storage racks. Note that in some instances, the reactivity of a fuel assembly will initially increase with burnup, then decrease with higher levels of burnup.

case, a dry storage technology has been employed. As of September 1998, installations of this kind were licensed at 11 nuclear plant sites.[16]

4. Experience with Administrative Controls

Since administrative controls were introduced to prevent criticality in fuel pools in the United States, there has been no documented criticality event in a pool. However, there have been numerous failures of relevant administrative controls. Appendix B describes some of those failures. The historical record summarized in Appendix B is almost certainly incomplete. Thus, the true record of failures of administrative controls is more severe, in terms of the potential for a criticality event, than Appendix B indicates.

Appendix B shows that fuel assemblies have been mispositioned on a number of occasions, resulting in the violation of burnup and/or enrichment limits. At Oyster Creek in 1987, a total of 184 fresh fuel assemblies were mispositioned in a pool.

Appendix B also describes incidents of error in criticality calculations, and incidents where the concentration of soluble boron in fuel pool water has been improperly managed. At McGuire Unit 1 in 1994, the soluble boron concentration dropped from 2,105 ppm to 1,957 ppm (a 7 percent drop), violating the plant's Technical Specifications.

Some of the incidents described in Appendix B involve failures of administrative controls that are not directly intended to prevent criticality. These failures show that administrative controls, as a class of safety measures, can and do fail.

Neither the NRC Staff nor any other entity has compiled a systematic database on the failure of administrative controls that are relevant to criticality prevention. In the

[16] *See* NRC Information Digest: 1998 Edition, NUREG-1350, Volume 10, Appendix H (November 1998).

absence of such a database, one cannot provide historically-based estimates of the likelihood of failure of relevant administrative controls.

5. The Millstone Unit 3 Refueling Outage #6

Millstone Unit 3 has undergone six refueling outages ("RFO's") since startup in 1986. The most recent was from May 11 to June 4, 1999. While the view was expressed by a member of the Board during the December 13, 1999 prehearing conference that Millstone's plague of problems involving systemic mismanagement was behind it, the record of RFO 6 strongly suggests otherwise.

Attached hereto as Exhibit 5 are the Reactor Engineering logs of RFO 6.[17] The Intervenors have distilled excerpts from the logs to highlight numerous irregularities during RFO 6 fuel movement activities. The excerpts appear as Exhibit 6.

May 15, 1999 was a particularly eventful day at Millstone Unit 3. May 15 entries include the following:

> 1917 SPF (Spent Fuel Pool") upender[18] will not lower. SM[19] granted permission to bypass.
>
> 2043 SFP upender will not lower. SM granted permission to bypass.
>
> 2116 SFP upender will not lower. SM granted permission to bypass interlock.[20]

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[17] The Intervenors requested that NNECO produce all reactor engineering logs of refueling outages at Millstone Units 1, 2 and 3 in their Third Set of Interrogatories and Requests for Production dated May 18, 2000. NNECO objected to production of *any* logs in its Motion for Protective Order dated May 22, 2000. The Board overruled the objection as to Millstone Unit 3 only. Thereafter, NNECO made available for inspection the original Reactor Engineering Logs for Unit 3 refueling outages 1 through 6.

[18] The upender is used to raise/lower fuel assemblies from the vertical position to/from the horizontal position so that they can be transported through the containment wall.

[19] "SM"stands for Shift Manager.

[20] The interlock is a design feature controlling the upender. NNECO lesson plan FHSO34T, Fuel Handling System, pages 20-30, describes the fuel transfer system. Various interlock bypasses are discussed. In each case, the use of the interlock bypass is described as being other than routine operation. For example, on

2209 SFP upender will not lower. SM granted permission to bypass overlock.

2245 SFP upender will not lower. SM granted permission to bypass interlock.

2302 SFP upender will not lower. SM gave permission to bypass.

2334 SFP upender will not lower. SM gave permission to bypass.

Given that the interlock bypasses are described as providing emergency override capability, it would appear that Millstone Unit 3 suffered seven (7) emergencies during a 4 hour and 17 minute period on May 15. It could be that NNECO personnel were abusing the interlock bypass capability. Instead of fixing the problem, they routinely bypassed the "an emergency override" of safety interlocks.

According to the log entries, permission from the Shift Manager was sought and obtained each time the emergency overrides were performed. The Shift Manager authorization is an administrative control. This evidence suggests that the administrative control was not successful in preventing recurring emergency conditions at Millstone Unit 3.[21]

It may be recalled that NNECO's Millstone problems surfaced when George Galatis discovered that NNECO was routinely performing full core offloads on Millstone Unit 1. The full core offload capability was described in the Updated Final Safety Analysis Report for Millstone Unit 1 as an emergency or abnormal occurrence, yet it was practiced almost every refueling outage at Millstone Unit 1.

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page 22 of 89, the CRANE INTERLOCK BYPASS is said to provide "an emergency override." Similarly, on page 23 of 89, the VALVE INTERLOCK BYPASS is said to provide "an emergency override." On page 25 of 89, the TRAVERSE INTERLOCK is described as "emergency override."

[21] "The Daily Scorecard: Millstone Megawatts vs. Outage Barriers: All the facts, stats and at-bats for Unit 3's Refueling Outage," was distributed internally at Millstone in May1999. On its face, the sheet suggests a less-than-serious attitude toward the refueling process and its hazards. It appears as Exhibit 7.

Five years later, NNECO still conducts fuel movements on Millstone Unit 3 under what are considered emergency conditions, elevating safety risks to its employees and to the public. Administrative controls are supposed to prevent emergencies – not facilitate them.

The Unit 3 SFP upender continued to malfunction after May 15, as log entries illustrate:

May 16:

0005 SFP upender will not go down. SM gave permission to bypass.

0128 SFP will not go down. SM gave permission to bypass.

0204 SFP upender will not go down. SM gave permission to bypass.

0430 SFP upender will not go down. SM gave permission to bypass.

0738 SFP upender could not go down to horizontal. Permission granted from shift manager to use bypass key. Upender lowered and taken out of bypass.

1344 Bypass key used to lower SFP upender.

1410 Bypass key used to lower SFP upender.

1427 Bypass key used to lower SFP upender.

1540 SFP upender would not raise with F/A[22] H37.

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[22] "F/A" refers to fuel assembly.

By May 17, problems with the SFP upender had not been resolved, as log entries illustrate:

1657 Bypass key utilized to lower upender at SFP transfer canal. Note. SM (Steve Lawhead) has given permission to the lead RE[23] to allow bypass of SFP upender (IAW OP 3303C Step 3.2.2) without checking in with him each time. This may change when the next SM comes on.

The upender problems continued on May 18:

1649 Upender will not raise. Contains F/A H44.

1720 Upender will not raise.

1835 Recommended standdown until troubleshooting of transfer system is complete and cause of upender problem is understood.

Problems with the SFP upender continued on May 19:

May 19: Received permission from Ray Martin to raise upender in SFP using bypass since it would not raise normally. Had cart run to full travel limit but would not raise. Bypassed interlock but frame still would not raise.

0130 Upender in SFP still unable to raise.

1523 Bypass key required to lower upender frame in spent fuel pool pit.

1720 Will not be picking up Fuel Assembly H-04 in the core until we get someone to access the upender problems. Getting progressively worse.

[23] "RE" refers to reactor engineer.

On June 4, the Shift Manager again granted permission to use the bypass key to lower the upender frame in the spent fuel pool at 1123.

Thus, the RE logs show no fewer than seventeen (17) instances where bypasses provided emergency override during RFO 6. The logs also show many other problems in fuel movements during RFO6.

For example, bypass keys required for performance of the emergency bypasses were not properly logged out or returned:

May 21:

0130 Removed bypass key #50 and 59 from the SPF area. Logged the area out of the FME area, returned the keys to the control room. The keys had not been properly logged out of the control room.

June 1:

Spent fuel bridge bypass key #59 is signed out to John S. This key is to be in spent fuel RE's possession.

Other problems during RFO 6 included lost communications. For example:

May 14:

1155 Loss of communications between CR and all stations.

1215 Communications lost – all Ericksons system went down

May 16: 1225 Ericson communications lost ~ 1 minute.

June 3:

Lost communications, apparently due to Erickson phone network problem.

2150 Gave up on communication – stopped fuel movement.

June 4:

0100 Still no communications.[24]

On June 3, the shuffleworks[25] connection lost power.

On May 16, the spent fuel pool bridge crane failed. (0802, 1640)

On May 18, a black tie wrap was found on the SFP transfer canal and had to be retrieved. (2003) On May 26, a wire nut was dropped into the spent fuel pool. (1000)

This recitation of RFO 6 irregularities does not even address the disastrous performance of the SIGMA system[26] which had recurring malfunctions throughout RFO 6 and caused frequent delays and work stoppages.[27]

NNECO produced its "Nuclear Oversight Audit Report" of Refueling Outage 6 during discovery. The executive summary of the audit report concludes that while "the Sigma machine experienced frequent malfunctions as did the Fuel Transfer System," "**The malfunctions were properly addressed by the refueling personnel**." (Emphasis added.)[28]

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[24] "CR" refers to control room.

[25] "Shuffleworks" refers to a fuel movement system.

[26] "SIGMA" refers to a fuel movement machine in containment.

[27] With so many SIGMA malfunctions, by June 3, log entries concluded that SIGMA had " lost its brain (again)."

[28] The Executive Summary is attached hereto as Exhibit 8.

However, a Condition Report ("CR") assessing the "Adverse Trend in Performance of the Refueling Equipment" during RFO 6 is less sanguine. The CR states as follows:

> "During core off load and core reload there were frequent equipment problems with the SIGMA refueling machine, the fuel transfer cart system, the primary communication system, and one failure of the spent fuel bridge crane. **These malfunctions affected the efficiency of the refueling operations and potentially challenged the safe handling of the fuel. Had the equipment failed in a manner such that a fuel assembly could have been damaged or been unable to be moved to a safe location, severe challenges to nuclear fuel safety could have occurred.**"[29]

The CR further states:

> "Upgrading the equipment to resolve performance problems is usually expensive and also requires significant time and effort by many departments.[30] The need to upgrade some of the equipment and improve the preventive maintenance program has been reinforced by the poor performance of this equipment in RFO6."[31]

The Condition Report is an alarming document.

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[29] The Condition Report, CR-M3-2236 ("Adverse Trend in Performance of the Refueling Equipment"), appears annexed hereto as Exhibit 9.

[30] Citing costs, NNECO may have delayed replacing lightbulbs in the spent fuel pool for as many as four or five years. See Deposition of Michael C. Jensen, Transcript, Exhibit 10, pages 82-83.

[31] Concerns about the expense of maintaining equipment has given rise to concern by some that under deregulation, economic pressures will lead to lessened preventive maintenance being done at nuclear power plants. See Matthew L. Wald, The New York Times, "Con Ed Put Off Plant Upgrade Over Rate Fear," June 30, 2000, page B1, Exhibit 11.

Equally damning is the memorandum of J.F. Beaupre, Unit 3 Technical Support Engineering , to D.E. Anderson, of NNECO nuclear oversight, of June 24, 1999.[32]

The Beaupre memorandum, a response to the CR, faults NNECO management for failing to perceive failures of the fuel handling system to be safety-significant.

The CR recommends corrective action to assure that the RFO 6 problems do not recur during RFO 7. However, the Beaupre memorandum identifies how many of the problems which marked RFO 6 derived from equipment deficiencies well known to plant operators prior to RFO 6, yet which had not been corrected After identifying the most significant equipment failures and repairs during RFO 6, the Beaupre memorandum identifies the apparent causes of the equipment failures as follows:

> " Apparent Causes
>
> 1. Corrective actions to resolve previously-identified fuel handling system equipment problems are frequently ineffective. The SIGMA control problems were identified in RO4, yet an EWR to upgrade the control system was not scheduled for implementation until Cycle 7. When the SIGMA cable supplied with a mast modification was identified as being too short, an effort to replace the cable with the proper length should have been initiated. An EWR to replace the spent fuel bridge hoist manual chain drive with a simpler design was approved, but the design change was given low priority and not completed prior to RFO6. The transfer cart holddown latch was modified after RFO1, yet failed to operate properly during RFO5 and RGO6. Efforts to repair the latch during RFO5 were unsuccessful. The new transfer cart holddown latch springs appear to be too weak to overcome friction in the latch bushing and return the latch to center. The transfer tube gate valve reach rod had slipped down during RFO5 and a modification to the support was not fully effective. Problems with

[32] A copy of the Beaupre memorandum appears as Exhibit 12.

the communications system were identified in RFO 5 and were not effectively resolved prior to RFO 6.

2. Operating experience at other plants is not effectively evaluated for applicability at Unit 3 and incorporated into the preventative maintenance program. Fuel handling system vendor manuals state that the equipment was designed to be reliable and the manuals specify the maintenance that needs to be performed prior to refueling outages. However, experience has shown that performing the minimum recommended maintenance does not assure good performance. As the equipment ages, unanticipated failures have occurred. Thoroughly reviewing fuel handling system problems that have occurred at other plants provides a foundation for evaluating the adequacies of Unit 3's PM program.

3. Preparing the fuel handling system for refueling is given low priority while the plaint is online. Preventive maintenance which is scheduled months before the outage is frequently deferred to a later start date because of other priorities. This results in significant pressure to complete the fuel handling system PMs in a short time, immediately prior to outage. The consequences of delaying the PMs is that problems identified must be corrected quickly and this sometimes results in the ineffective corrective actions previously identified."

The Beaupre memorandum closes with a scathing criticism of Millstone nuclear oversight for what is termed its failure to understand the safety significance of fuel handling system equipment and its failure to communicate such to NNECO management:

"4. Failures of fuel handling system equipment that delay refueling are not perceived to be safety-significant. This is demonstrated by the EWR prioritization process that assigns point values to EWRs based on significance (i.e., safety, cost-savings, ALARA, etc.) **A review of EWRs related to the reliability of the fuel handling equipment shows that the safety significance of equipment**

upgrades is not fully understood and communicated to management."
(Emphasis added)

The record of Millstone Unit 3 RFO 6 demonstrates an alarming pattern of failure to properly maintain equipment, reliance on emergency administrative controls and disregard for past plant system deficiencies, exposing the employees and public to heightened risk of harm and, in the terminology of the CR, potentially posing a "severe challenge to nuclear fuel safety."

The NRC Staff completed an inspection of Unit 3 on June 14 which encompassed the RFO 6 outage and it submitted a report by James C. Linville, Acting Director, Millstone Inspection Staff, Office of the Regional Administrator, dated July 9, 2000, addressed to R.P. Necci, NNECO Vice President for Nuclear Oversight and Regulatory Affairs.[33]

The NRC Staff member's letter characterizes NNECO's management of RFO 6 outage as follows:

> "Refueling outage activities were in progress at Unit 3 during most of this inspection period. We observed that the challenges that were encountered during RFO6 were methodically evaluated and appropriately dispositioned by your staff using a team approach."

[33] The cover letter appears as Exhibit 13.

B. The Millstone License Amendment Application

By letter dated March 19, 1999, Northeast Nuclear Energy Company (NNECO) submitted an amendment request for the Millstone Unit 3 operating license, seeking to expand the storage capacity at the facility. Attachment 3 to that application described the company's expansion proposal. Millstone Unit 3 currently has 21 free-standing spent fuel racks with a total storage capacity of 756 fuel assemblies. Each existing rack is a 6x6 array using Boraflex as the neutron absorption material.

The existing Technical Specifications for Millstone Unit 3 divide the presently installed racks into two regions. Per Technical Specification Definition 1.40, the Region I racks use a 3-out-of-4 configuration with a fuel cell blocker in the fourth location. Per Technical Specification Definition 1.41, the Region II racks do not have fuel cell blockers. Technical Specification Surveillance Requirement 4.9.13.1 controls placement of fuel in Region I and II. When the fuel assembly enrichment and burnup parameters are to the right of the line drawn on Technical Specification Figure 3.9-1, a fuel assembly cannot be stored in a Region II rack.

NNECO seeks permission to install up to fifteen (15) additional racks in the spent fuel pool at Millstone Unit 3. Five (5) of the proposed new racks will be 7x10 arrays using Boral as the neutron absorption material. NNECO proposes to designate these five new racks as Region 1 of the spent fuel pool. The company seeks to use Region 1 to store fuel assemblies with a nominal 5.0 w/o U-235 enrichment in a 3-out-of-4 configuration without burnup restrictions. In the 3-out-of-4 configuration, a fuel cell blocker is proposed for criticality control. The application also provides for fuel assemblies to be stored in Region 1 in a 4-out-of-4 configuration (i.e., no cell blockers) when restrictions are placed on burnup and enrichment.

The remaining ten (10) proposed new racks have varying array dimensions using Boral as the neutron absorption material. NNECO proposes to designate these ten new racks as Region 2 of the spent fuel pool. The application provides for fuel assemblies to be stored in Region 2 in a 4-out-of-4 configuration (i.e., no cell blockers) with restrictions placed on burnup and enrichment. These restrictions are more restrictive than those imposed on storage in Region 1 racks.

NNECO proposes to re-designate the 21 existing racks as Region 3 of the spent fuel pool. The application provides for fuel assemblies to be stored in Region 3 with more restrictions on burnup and enrichment than imposed on the Region 2 (and 1) racks. In addition, the application provides for credit to be taken for the decay of fissile plutonium and the buildup of americium over time for the irradiated fuel stored in the Region 3 racks.

C. Connecticut Coalition Against Millstone/Long Island Coalition Against Millstone Intervention in the Licensing Proceeding

On September 7, 1999, the NRC published notice of opportunity for a hearing on the proposed license amendment, 64 Fed. Reg. 48672. The Intervenors, Connecticut Coalition Against Millstone (CCAM) and Long Island Coalition Against Millstone (CAM), filed a timely request for hearing and intervention petition on October 6, 1999.

CCAM is an organization of environmental and safe-energy organizations and individuals based in Mystic, Connecticut. CAM is an organization of environmental and safe-energy organizations and individuals based in East Hampton, New York. On November 17, 1999, in their Supplemental Petition to Intervene, the Intervenors submitted contentions challenging the adequacy of the License Amendment Application.

The Licensing Board conducted a prehearing conference on December 13, 1999 in New London, Connecticut. At the conclusion of the conference, the Licensing Board ruled that both CCAM and CAM have standing to intervene in these proceedings.

In its Prehearing Conference Order issued on February 9, 2000, the Licensing Board admitted three of the Intervenors' contentions, Nos. 4, 5 and 6. As admitted by the Licensing Board, Contention 4 reads as follows:

Contention 4: Undue and Unnecessary Risk to Worker and Public Health and Safety

The new set of administrative controls trades reliance on physical protection for administrative controls to an extent that poses an undue and unnecessary risk of a criticality accident, particularly due to the fact that the licensee has a history of not being able to adhere to administrative controls with respect, inter alia, to spent fuel pool configuration.

Contention 5 reads as follows:

Contention 5: Significant Increase in Probability of Criticality Accident

NNECO proposes to eliminate an existing barrier against criticality in the spent fuel pool at Millstone Unit 3. The present Technical Specifications require soluble boron to be maintained in the spent fuel pool's water at all times. NNECO proposes to change the requirement for soluble boron as follows:

The proposed Technical Specifications will require a minimum concentration of 800 ppm [parts per million] of soluble boron in the pool water during fuel movement to assure keffective will remain less than or equal to 0.95 assuming a dropped or misloaded fuel assembly. The surveillance interval or this soluble boron concentration in the proposed Technical Specifications is consistent with Westinghouse improved STS [Standard Technical Specifications] 3.7.16.[34]

The petitioners contend that NNECO's proposal presents a significant increase in the probability of a criticality accident.

Basis: The present Technical Specifications for Millstone Unit 3 require soluble boron to be maintained within the spent fuel pool water any time irradiated fuel assemblies are stored in the pool. The proposed change would require soluble boron to be maintained only during fuel movements – not at times between fuel movements while irradiated fuel assemblies are stored in the pool. The evaluation submitted by NNECO clearly stated that a single fuel movement error, which is a credible event within the plant's design and licensing bases, can result in the required criticality margin being violated unless there is soluble boron in the spent fuel pool water.

The inadvertent misplacement of fresh fuel assembly has the potential for exceeding the limiting activity, should there be a concurrent and independent condition resulting in the loss of all soluble poison. Assuming the presence of soluble poison during fuel handling operations will preclude the possibility of the simultaneous occurrence of the two independent accident conditions. The largest reactivity increase would occur if a fresh fuel assembly of 5.0 wt%235U enrichment were to be inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. Under this accident condition, credit for the presence of soluble poison is permitted by the NRC guidelines. Calculations indicate that 800 ppm soluble boron, that is to be required by the Technical Specifications during fuel handling operations, is more than adequate to assure that the limiting keffective of 0.945 is not exceeded.[35]

And

[34] Page 5 of Attachment 3 to the Application dated March 19, 1999.
[35] Page 4-8 of Attachment 5 to the Application dated March 19, 1999.

With the assumption that Boraflex panels are replaced by water, the moderator temperature coefficient of reactivity in Region 3 is positive. Therefore, an increase in spent fuel pool temperature above the normal operating conditions (i.e., above 160 F) has the potential for exceeding the limiting reactivity in Region 3, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Calculations indicate that 100 ppm soluble boron is more than adequate to assure that the limiting keffective of 0.945 is not exceeded or temperatures greater than 160 F and boiling.[36]

If the Technical Specifications for Millstone Unit 3 are changed as requested by NNECO, it is credible that a human error could result in the wrong fuel assembly being loaded unto a Region 3 rack. That such an error is credible is implicitly conceded by NNECO's evaluation of such an event. With the soluble boron concentrations required by the "revised" Technical Specifications during fuel movements, this loading error would not cause a criticality.

But, once the fuel movements are stopped, the "revised" Technical Specifications no longer require soluble boron to be maintained in the spent fuel water. If the misloaded fuel assembly remains undetected and the soluble boron concentration drops, a criticality could occur.

The NRC's records include reports of misloaded fuel assemblies remaining undetected for long periods of time.[37]

NNECO's application posits that a misloading error may be made in the Millstone Unit 3 spent fuel pool. NNECO's evaluation of such a misloading error determined that a configuration which could yield criticality if it were not for the soluble boron in the water. Yet, NNECO proposes to remove the soluble boron Technical Specification without at least providing a surveillance requirement to check for misloaded fuel assemblies at the termination of fuel movements.

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[36] Page 4-9 of Attachment to the Application dated March 19, 1999.

[37] Stewart D. Ebneter, Regional Administrator, NRC, to J.W. Hampton, Vice President-Oconee Site, Duke Power Company, "Notice of Violation and Proposed Imposition of Civil Penalty - $50,000 (NRC Inspection Report Nos. 50-269/96-02, 50-270/96-02 and 50-287/96-02),"March 5, 1996, and M.E. Reddemann, General Manager – Hope Creek Operations, Pubic Service Electric & Gas Company, to NRC,

Contention 6 reads as follows:

Contention 6: <u>Proposed Criticality Control Measure Would Violate NRC Regulations</u>

__The criticality control measures proposed by NNECO would violate Criterion 62 o the General Design Criteria (GDC) set forth in Part 50, Appendix A. GDC 62 requires that: "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." In violation of this requirement, NNECO proposes to seek to prevent criticality at Millstone Unit 3 by the use of ongoing administrative measures.

Basis: GDC 62 is the sole regulatory foundation or criticality control in fuel pools. The NRC staff has employed other documents in its consideration of criticality, but these documents are not regulations. For example, the NRC has repeatedly referred to a Draft or Comment o Proposed revision 2 to regulatory Guide 1.13, dated December 1981, titled, "Spent Fuel Storage Facility design Basis." That document, in addition to being a draft, is not a regulation.

The NRC staff has on various occasions allowed nuclear power plant licensees to rely upon administrative measures for criticality control, as NNECO proposes in this application. Such reliance violates GDC 62, and therefore violates NRC regulations. NNECO proposes to rely upon the following administrative measures of criticality control at Millstone Unit 3: (1) maintenance of a given content o soluble boron in pool water; (2) limits on fuel enrichment/fuel burnup in Region 1 4-out-o-4 racks and Region 2 racks; and (3) limits on fuel enrichment/fuel burnup and fuel decay time in Region 3 racks.

GDC 62 requires reliance on physical systems or processes, rather than administrative measures, under both normal conditions and accident conditions. For practical application o GDC 62, a "credible" range of accident conditions must be defined. The NRC has not formally provided such a definition. A potentially useful definition of credible accident conditions is provided, by implication in draft regulatory Guide 1.13, cited above.

"Hope Creek Generating Station Licensee Event Report No. 95-042-00," March 25, 1996, and Tennessee Valley Authority, Licensee Event Report No. 50-260/80-037-01, October 9, 1980.

Paragraph 1.4 of Appendix A of Draft Regulatory Guide 1.13 states; all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could NOT occur without at least two unlikely, independent and concurrent failures or operating limit violations." This statement could be interpreted as saying that the set of non-credible accident scenarios, for the purposes of criticality control, encompasses scenarios involving at least two unlikely, independent and concurrent failures or violations. All other accident scenarios would then be regarded as credible.

Experience at U.S. nuclear power plants shows that failure of administrative measures that seek to limit fuel enrichment/fuel burnup or fuel decay time is a likely occurrence. Moreover, it is likely that these administrative measures will fail in such a manner that more than one fuel assembly is out of compliance with specified limits. Also, failure of administrative measures that seek to limit fuel enrichment/fuel burnup or fuel decay time can precede or follow, rather than being concurrent with, failure of administrative measures that seek to maintain a given content of soluble boron in pool water. As a result, if the Millstone Unit 3 fuel pool were to operate as NNECO proposes in this application, a variety of accident scenarios involving criticality could occur that are credible according to the definition of credibility which is implied by Paragraph 1.4 of Appendix A of Draft Regulatory Guide 1.13. Thus, GDC 62 would be violated under accident conditions.

As required by 10 C.F.R. Section 2.1111, the Board offered the parties an opportunity to invoke the hybrid hearing process outlined in Subpart K. This process establishes a 90-day discovery period, followed by the filing of a detailed written summary of all facts, data and arguments that each party intends to rely on to support the existence of a genuine and substantial dispute of fact regarding any admitted contentions. Following this filing, an oral arguments held. 10 C.F.R. Section 1113. NNECO invoked the hybrid hearing process, and therefore this Summary is being filed herewith.

During the pendency of these proceedings, on March 19, 1999, NNECO submitted to the NRC proposed changes to selected Technical Specifications whereby the proposed boron concentration of 800 ppm was to be maintained whenever fuel is stored in the spent fuel pool. NNECO's proposal specifically referenced the instant proceedings and appeared to be directly responsive to Intervenors' Contention 5.[38]

The parties duly conducted discovery pursuant to 10 C.F.R. Section 2.1111. On May 18, 2000, the Intervenors filed their Third Set of Interrogatories and Request for Production directed to NNECO. On May 22, 2000, NNECO filed a Motion for Protective Order objecting to the Intervenors' May 18, 2000 discovery request. On May 25, 2000, the Board served notice of a telephone conference to be conducted on May 26, 2000 to address discovery and other related matters. The Board rendered rulings on the motion and objection during the conference, as later memorialized in the Board's Memorandum and Order issued on June 8, 2000. While overruling certain of NNECO's objections to the discovery, the Board granted a protective order regarding others. Most particularly, the Board declined to order NNECO to respond to Interrogatory A2 (parts 3 through 7), which sought information regarding boron dilution.
39

Intervenors' Interrogatory A2 was submitted following the discovery site inspection conducted at the spent fuel pool on May 10, 2000, during which numerous piping systems were observed suspended from the ceiling and along the walls of the spent fuel pool room.[40] The presence of a roof drain above the pool was noted. The piping systems could potentially add water to the pool, causing a dilution of soluble boron.

In its discovery ruling, the Board also declined to order NNECO to respond to Intervenors' Interrogatory A4 in terms they had requested. Interrogatory A4 states as follows:

A4 Calculations of K-Eff

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[38]R.P Necci letter to U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification – Spent Fuel Pool Rerack (TSCR 3-22-98), dated April 17, 2000.

[39]A copy of Interrogatory A2 appears as Exhibit 14.

[40] A set of photographs depicting various views within the spent fuel pool building, including the presence of piping systems, appears annexed hereto as Exhibit 15. They were taken by NNECO during the site inspection on May 10, 2000 at the request of the Intervenors.

Given the implementation of the proposed re-racking of the Millstone 3 pool, and assuming an absence of soluble boron, what would be the calculated k-effective in each of the regions of the pool if various combinations of fresh fuel assemblies were placed in the racks? For this purpose, various combinations of fresh fuel assemblies would include one assembly, two adjacent assemblies, four adjacent assemblies, and a full rack, where in each case the surrounding cells would be occupied by assemblies of the highest reactivity allowed by the Technical Specifications.

NNECO told the Board it would include various k-effective scenarios in its written testimony, but it was reluctant to reveal its calculations prior to its written submission.

The Board left it to NNECO's discretion to furnish to the Intervenors its k-effective calculations as they "became available." NNECO has provided the results of some criticality calculations in its June 21, 2000 supplemental submission.

IV. NEW ADMINISTRATIVE CONTROLS POSE AN UNDUE AND UNNECESSARY RISK OF A CRITICALITY ACCIDENT

NNECO's license amendment application proposes the introduction of new administrative controls for the prevention of criticality. This proposed change in the operation of the Millstone Unit 3 fuel pool would significantly increase the probability that a criticality accident or a violation of criticality limits will occur in the pool.[41] The increased probability of a criticality accident would create an undue increase in the burden of risk that arises from operation of the fuel pool. Neither the applicant nor the NRC Staff has a proper understanding of the burden of risk implied by the license amendment, and neither has performed the technical analysis necessary to establish such an understanding. The lack of appropriate analysis provides sufficient grounds for

[41] Throughout this Summary, the phrase "a criticality accident" should be generally understood to mean "a criticality accident or a violation of criticality limits." Criticality limits are intended to establish a factor of safety against criticality accidents.

rejection of the license amendment application. Absent such a rejection, the lack of appropriate analysis provides grounds for the conduct of a full evidentiary hearing, in part because the response to CCAM/CAM's discovery was inadequate. Finally, the increased burden of risk would be unnecessary, because alternative modes of spent fuel storage -- especially dry storage, using well-established technology -- are available, allowing the applicant to increase its stock of spent fuel without increasing the risk that arises from operation of the Millstone Unit 3 fuel pool.

A. Significant Increase in the Probability of a Criticality Accident

The proposed license amendment would significantly increase the probability of a criticality accident at the Millstone Unit 3 pool, through the interaction of five factors. First, the amendment would lead to increased complexity of the administrative controls upon which NNECO will rely to prevent a criticality accident. Second, failure of administrative controls can lead to a criticality accident, and a failure of this type is more likely if administrative controls are more complex. Third, criticality calculations can contain errors, and reliance on administrative controls of increased complexity will increase the potential that such errors will lead to a criticality accident. Fourth, experience shows that administrative controls on fuel positioning are likely to fail, and failure is more likely if these administrative controls are more complex. Fifth, there is a significant probability that the concentration of soluble boron in the pool water will be insufficient to prevent a criticality accident at the time of or subsequent to a fuel mispositioning event.

1. Increased Complexity of Administrative Controls

The proposed amendment would increase the number of fuel storage regions within the Millstone Unit 3 spent fuel pool by 100 percent. The current pool has two regions while the proposed pool configuration would feature four regions.[42]

The proposed amendment would increase the number of parameters affecting storage in the Millstone Unit 3 spent fuel pool by 50 percent. The current pool storage options are dependent on two parameters (enrichment and burnup) while the proposed pool storage options would rely on three parameters (enrichment, burnup, and decay time).

Table IV-1 indicates the storage options for various enrichment, burnup, and decay time combinations in the spent fuel pool as presently licensed and as sought by NNECO.

Table IV-1
Allowable Positioning of Fuel in the Millstone Unit 3 Fuel Pool

Enrichment (w/o U-235) / Burnup (GWD/MTU)	Existing License		Proposed License			
	Region I 3-out-of-4	Region II 4-out-of-4	Region 1 3-out-of-4	Region 1 4-out-of-4	Region 2	Region 3
3.7 / 0	Yes	Yes	Yes	Yes	**No**	**No**
3.7 / 10	Yes	Yes	Yes	Yes	**No**	**No**
3.7 / 20	Yes	Yes	Yes	Yes	Yes	**No**
3.7 / 30	Yes	Yes	Yes	Yes	Yes	***Maybe*** #
4.0 / 0	Yes	**No**	Yes	**No**	**No**	**No**

[42] The application refers to Regions 1, 2, and 3, but proposed Technical Specification 1.41 would permit fuel to be stored in Region 1 in either a 3-out-of-4 or a 4-out-of-4 configuration. This provision essentially creates four regions.

4.0 / 20	Yes	Yes	Yes	Yes	**No**	**No**
4.0 / 35	Yes	Yes	Yes	Yes	Yes	***Maybe***##
5.0 / 0	Yes	**No**	Yes	**No**	**No**	**No**
5.0 / 30	Yes	Yes	Yes	Yes	**No**	**No**
5.0 / 45	Yes	Yes	Yes	Yes	Yes	Maybe*

Notes to Table IV-1:

\# - Yes if decay time greater than 10 years, **No** otherwise.

\## - Yes if decay time greater than 5 years, **No** otherwise.

* - Yes if decay time greater than 20 years, **No** otherwise.

It is immediately evident from Table IV-1 that the proposed license amendment would significantly increase the complexity of administrative controls on the positioning of fuel in this pool. As a result, there would be significantly more opportunities for a fuel mispositioning event.

2. Failure of Administrative Controls Can Lead to a Criticality Accident

The evaluation submitted by NNECO in support of the amendment request stated that a fuel mispositioning event involving a single fuel assembly can result in the required criticality limit being violated unless there is soluble boron in the pool water:

> "The inadvertent misplacement of a fresh fuel assembly has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Assuring the presence of soluble poison during fuel handling operations will preclude the possibility of the simultaneous occurrence of the two independent accident

conditions. The largest reactivity increase would occur if a fresh fuel assembly of 5.0 wt % U235 enrichment were to be inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity. Under this accident condition, credit for the presence of soluble poison is permitted by the NRC guidelines. Calculations indicate that 800 ppm soluble boron, that is to be required by the Technical Specifications during fuel handling operations, is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded."[43]

and

"With the assumption that the Boraflex panels are replaced by water, the moderator temperature coefficient of reactivity in Region 3 is positive. Therefore, an increase in spent fuel pool temperature above the normal operating conditions (i.e., above 160 F), has the potential for exceeding the limiting reactivity in Region 3, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. ... Calculations indicate that 100 ppm soluble boron is more than adequate to assure that the limiting k_{eff} of 0.945 is not exceeded for temperatures greater than 160 F and boiling."[44]

Appendix C shows that a variety of failures of administrative controls will lead to a criticality accident or a violation of criticality limits. Failures of this type have occurred, and are more likely if administrative controls are more complex. Greater complexity of administrative controls creates more opportunities for their failure, as illustrated by Table IV-1.

3. Criticality Calculations Can Contain Errors

NNECO's above-cited evaluation is based upon a theoretical calculation of the neutron multiplication factor k_{eff}, to determine the pool's margin against criticality.

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[43] Page 4-8 of Attachment 5 to the Application dated March 19, 1999.
[44] Page 4-9 of Attachment 5 to the Application dated March 19, 1999.

Nuclear industry experience, as illustrated by Appendix B, demonstrates that criticality calculations of this kind can be non-conservative.

Examples of non-conservative criticality calculations include:

McGuire Units 1 and 2: March 2, 2000 (Licensee Event Report 369/00-03) (March 30, 2000)[45]

During fuel pool criticality calculations using new models, it was discovered on March 2, 2000 that non-conservative assumptions had been used in prior criticality calculations. As a result, the k_{eff} might exceed 0.95 for postulated off-normal conditions with 0 ppm boron concentration. The Technical Specifications require that fuel stored in the spent fuel pool must not exceed 0.95 k_{eff} when the pool is flooded with unborated water.

Criticality calculations for McGuire had been performed using 2-dimensional geometry. It had been believed that the 2-dimensional calculations were conservative enough to easily bound fuel assemblies stored in the spent fuel pools. However, the 3-dimensional calculations using non-uniform fuel assembly axial burnup distributions showed that the results from the 2-dimensional calculations were non-conservative for lower burnups and enrichments. The plant's licensee concluded that: "Given the actual fuel assembly burnups and the existing limits, the potential existed that k_{eff} would exceed 0.95 under the postulated unborated condition."

Thus, criticality calculations were performed for both units at the McGuire Nuclear Station for more than a decade using non-conservative assumptions that caused Technical Specifications to be violated.

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[45] A copy of this LER is attached as Exhibit 16.

Millstone Unit 2: February 14, 1992 (Licensee Event Report 336/92-003-01) (June 25, 1992)[46]

On February 14, 1992, it was discovered that a calculation error existed in the criticality analysis for the Region 1 spent fuel storage racks. The originally calculated value of K_{eff} was 0.922. The newly calculated value of K_{eff}, for the same conditions, was 0.963. This error arose from the use of two inappropriate assumptions in the calculations.

Millstone Unit 2: (NRC Information Notice 92-21, Supplement 1, Spent Fuel Pool Reactivity Calculations) (April 22, 1992)[47]

The NRC clarified the extent of errors made in the reactivity calculations performed by ABB Combustion Engineering for the Millstone Unit 2 spent fuel pool. The revised calculations showed that the absorption cross section in Boraflex for the epithermal energy group is significantly self-shielded; however, this was not accounted for in the original calculations. This oversight resulted in overestimating neutron absorption and a corresponding lower calculated k_{eff} in that region. This error was in addition to having used an inaccurate geometric buckling term.

These actual industry events show that criticality calculations can be non-conservative. One of the flawed calculations at Millstone Unit 2 resulted in the calculated criticality margin being 0.041 $\mathbf{K_{eff}}$ less than the actual margin. The Technical Specifications for Millstone Unit 3 only require a margin of 0.05 K_{eff}.

[46] A copy of this LER is attached as Exhibit 17.
[47] A copy of this Supplement is attached as Exhibit 18.

Reliance on administrative controls of increased complexity requires the performance of additional criticality calculations, involving a greater number of parameters. The number of criticality calculations can increase more than proportionally with the number of regions and parameters. For example, an increase in the number of regions in a pool, as NNECO proposes at Millstone Unit 3, requires new criticality calculations not only for each individual region but also for all combinations of interfacing zones where regions are adjacent. Assuming a fixed incidence of non-conservative errors in criticality calculations, increased complexity of administrative controls in a pool will lead to an increased number of non-conservative errors in those calculations and, in turn, to an increased potential that calculational errors will lead to a criticality accident in the pool.

4. Fuel Can be Mispositioned

Nuclear industry experience, as illustrated by Appendix B, is replete with failures of administrative controls on fuel positioning. Many of the resulting fuel mispositioning events have involved more than a single fuel assembly. Examples include:

> **Byron Station: May 28, 1996 (Licensee Event Report 454/96-008-00) (June 25, 1996)**[48]
>
> On May 28, 1996, three fuel assemblies were found to be present in Region 2 of the spent fuel pool without meeting Technical Specifications requirements. The assemblies did not meet the minimum burnup requirements, nor were they checkerboarded. The required (actual) burnups (in MW-days per tonne U) were: 32,651 (32,648); 32,651 (32,638); and 32,771 (32,728). Two of the three non-complying assemblies were placed in Region 2 in August 1993, and the third assembly was placed in Region 2 in January 1995.

In the period August-November 1994, Byron Station engineers had built a computer spreadsheet to calculate assembly compliance with criteria for placement in Region 2. This spreadsheet did not detect the non-compliance of the three assemblies, because the spreadsheet was loaded with incorrect data for the assemblies' initial enrichment, storage location, and burnup.

When first placed in Region 2, each of the three assemblies was in compliance with minimum burnup requirements as then calculated. Subsequent re-calculations led to increased minimum burnup requirements (operative in December 1994), which put the assemblies out of compliance. Although the degree of non-compliance was relatively small, it is significant that the non-compliance arose from faulty data entry and was not detected for a long period.

Farley Unit 1: March 23, 2000 (Licensee Event Report 348/2000-004-00) (April 20, 2000)[49]

While fuel location data were being incorporated into a Spent Nuclear Material tracking software package on March 23, 2000, it was discovered that three fuel assemblies were loaded in the spent fuel pool in a configuration contrary to the Technical Specifications. Specifically, the three assemblies had insufficient burnup for their storage locations. The loading error had first occurred during core offload ten days earlier.

The plant's licensee reported: "Manual verification of the acceptability of the proposed offload configuration on March 11, 2000 failed to identify that three assemblies had insufficient burnup for their planned storage locations." The licensee reported that the error occurred because: **"As the**

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[48] A copy of this LER is attached as Exhibit 19.
[49] A copy of this LER is attached as Exhibit 20.

SFP [spent fuel pool] approaches capacity with time, the complexity of the task of determining acceptable storage configuration has increased." [emphasis added]

Thus, Farley had a Technical Specification requirement allowing only fuel assemblies with higher burnup to be stored in certain storage locations. However, the complexity of the associated administrative controls caused that Technical Speciication requirement to be violated.

McGuire Unit 1: October 24, 1991 (Licensee Event Report 369/91-016-00) (November 25, 1991)[50]

Plant personnel discovered that 11 fuel assemblies had been stored in the spent fuel pool in a manner contrary to Technical Specifications requirements. These requirements stipulated that, if a checkerboard pattern was used in Region 2 for storage of fuel that would have been non-complying if not stored in a checkerboard pattern, then one row between normal storage locations and checkerboard locations would remain vacant. The requirement for a vacant row was not satisfied from March 23, 1990 through October 23, 1991. The licensee attributed this error to poorly written procedures.

It should also be noted that 9 of the 11 previously designated fuel assembly locations were changed on March 23, 1990 in order to maximize the number of open locations in anticipation of a core offload.

Oyster Creek Unit 1: January 21, 1987 (Licensee Event Report 219/87-006-00) (February 24, 1987)[51]

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[50] A copy of this LER is attached as Exhibit 21.
[51] A copy of this LER is attached as Exhibit 22.

On January 21, 1987 it was discovered that fresh fuel with an enrichment higher than the Technical Specification limit had been stored in the spent fuel pool, beginning on February 27, 1986. The Technical Specification limit on average planar enrichment was 3.01 wt% U-235.

A total of 204 fresh fuel assemblies, with an average planar enrichment of 3.19 wt% U-235, were received at the plant in 1986. The dry storage vault had a capacity for 140 assemblies. Thus, 64 fresh assemblies were initially stored in the spent fuel pool. As the refueling outage progressed, more assemblies were taken out of the dry storage vault, channelled, and stored in the spent fuel pool. Ultimately, 184 noncompliant fresh assemblies were stored in the spent fuel pool prior to the start of core reload in August 1986. By the time the core had been fully reloaded (on September 14, 1986), all of the fresh fuel had been removed from the spent fuel pool.

The licensee ascribed this occurrence to personnel error. Specifically, the plant's safety analysis did not take into account the possibility that fresh fuel would be stored in the spent fuel pool.

Fuel handling errors at Millstone were documented during discovery. Exhibit 23 contains the disclosures of NNECO in response to the Intervenors' First Set of Interrogatories dated March 21, 2000 (request for "all instances of error at Millstone in managing, moving, placing or tracking fresh or spent fuel at Millstone"). There may have been others.

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The incident on April 26, 1994, when a crane operator at Millstone Unit 3 lowered a fuel assembly into SFP N-7 instead of N-6, illustrates some of the common

[52] So NNECO's Michael C. Jensen acknowledged at his deposition. Exhibit 10, page 19.

circumstances which can and have led to fuel mispositionings.[53] The fuel assembly was lowered 6 inches before the error was noticed. The crane operator reported poor lighting conditions ("[D]ue to the poor lighting in that area, I did not see the fuel assembly. The PEO also checked, but he, apparently, did not see it either."); fatigue due to overwork ("Also I've been up since 0130. I came in to work 0500."); distractions ("I was holding a conversation with Tom concerning mode zero alternate fuel pool cooling. I forgot to cross out the cell we had just loaded."), inadequate procedures ("The engineer should have a better way of keeping track of fuel assemblies.") and confusing procedures ("Some confusion may be created by the number of procedures in use.") After the incident, the crane operator recognized that he should have notified the shift supervisor when the misplacement occurred and fuel movement should have been halted.[54] According to the Reactor Engineering log of the incident, the crane operator "explained that it would be easier if we had bigger numbers on the bridge. He was also distracted a little when everyone was talking about the upcoming move problems with R-17 and R-18."[55]

Failure of administrative controls on fuel positioning are more likely if the administrative controls are more complex. Greater complexity creates more opportunities for administrative controls to fail. Thus, NNECO's proposed license amendment would significantly increase the probability of fuel mispositioning at the Millstone Unit 3 pool.

5. Dilution of Soluble Boron Can Occur

Nuclear industry experience, as illustrated by Appendix B, shows that the soluble boron in the water in fuel pools can be diluted. Moreover, at the Millstone Unit 3 pool there are numerous systems and mechanisms that could remove water from the pool and add unborated water, thus leading to a significant probability of a dilution of soluble

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[53] The incident is one of the reported events documented in Exhibit 23; it was discussed during the Jensen deposition. See Exhibit 10, pages 60-66.
[54] Exhibit 10, page 63.
[55] A copy of pages 48-48 of the Reactor Engineering log RFO 5 is annexed as Exhibit 24.

boron in the pool water. [56] Section B, below, addresses the availability of technical analysis that pertains to this probability.

Appendix C shows that a variety of combinations of fuel mispositioning and boron dilution events at the Millstone Unit 3 pool would lead to a criticality accident. NNECO's proposed license amendment would significantly increase the probability of fuel mispositioning in this pool and, therefore, the probability of a criticality accident.

B. Undue Increase in the Burden of Risk

Operation of the Millstone Unit 3 fuel pool places a burden of risk on members of the workforce and the surrounding public. That burden of risk has not been properly characterized by NNECO or the NRC Staff, either for present conditions or for the conditions that would arise after the proposed license amendment. The lack of proper characterization is itself a part of the burden of risk, because it promotes uncertainty and concern on the part of potentially affected people.

1. Increased Probability of a Criticality Accident Would Increase the Burden of Risk from Pool Operation

One component of the present burden of risk is attributable to the potential for a criticality accident at the pool. That component would increase significantly after the proposed license amendment, because the amendment would cause a significant increase in the probability of a criticality accident.

As indicated in Appendix C, a criticality accident could release sufficient energy to damage the affected fuel assemblies. Significant onsite and offsite radiation exposures are potential outcomes of the accident.

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[56] Robert Griffin, Millstone Chemistry Manager, during his May 12, 2000, deposition, described an incident at Millstone Unit 2 when hydraulic fluid entered the spent fuel pool. The liquid contents of the pool were put through an osmotic process for purification. Griffin Deposition Transcript, Exhibit 25, page 128.

2. Neither the Applicant nor the NRC Staff has a Proper Understanding of the Increased Burden of Risk

Neither NNECO nor the Staff maintain or have access to a systematic database of experience with failures of administrative controls that are relevant to prevention of criticality. Therefore, neither party can provide historically-based estimates of the likelihood of failure of relevant administrative controls. Also, neither party has determined the combinations of fuel mispositioning and soluble boron dilution events that constitute the envelope of criticality for this pool.[57] Thus, neither party can provide a reliable estimate of the probability of a criticality accident at the pool, either for present conditions or for post-amendment conditions. Finally, neither party has estimated the consequences of a criticality accident.

NNECO's proposed license amendment at Millstone Unit 3 represents a relaxation of standards for criticality prevention, because the amendment requires reliance on administrative controls of increased complexity. Both NNECO and the Staff are content with this situation, although neither can provide an estimate of the amendment's implications for the probability and consequences of a criticality accident.

The Staff compounds this situation by its marked lack of technical curiosity and independent investigation regarding the implications of the proposed license amendment. This phenomenon was manifested during discovery, commencing when the Intervenors propounded interrogatories to the NRC staff. (Intervenors' First Set of Interrogatories and Requests for Production Directed to NRC dated March 22, 2000) The NRC staff selected Anthony C. Attard, Ph.D., as one of three designated affiants charged with providing information under oath responsive to the Intervenors' discovery requests.[58]

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[57] The envelope of criticality is defined in Appendix C.

[58] See "NRC Staff's Response to Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone's First Set of Interrogatories and Requests for Production Directed to U.S. Nuclear Regulatory Commission" dated April 10, 2000.

When the Intervenors later deposed Dr. Attard on May 11, 2000, he stated that he had been assigned the task of reviewing NNECO's License Amendment Application to rerack in the Millstone Unit 3 spent fuel pool.[59] However, he acknowledged that he had no previous experience in reviewing a license amendment to expand storage capacity in a spent fuel pool.[60]

Dr. Attard was asked to identify the standards the application was required to meet. He replied as follows:

"Well, again, there was – it was kind of done for me in a way. . . . I was kind of presented in the front of the Holtec report. For example, you know, they list a series of – I think it's Reg. Guide 11- 1.213, the so-called Grimes letters, Grimes letter, Dr. Kopp's letter. There's about five or six bullets in there that talks about the – what they had to meet. And I would have went to the same thing."[61]

Dr. Attard was asked if he had determined whether the application satisfied the standards Holtec had identified. The following colloquy ensued:

A I have not yet.

Q What else do you need to look at?

A Well, I mean, I haven't looked at all of them is what I meant.

Q Haven't looked at what?

A At all those ones that I told you. I haven't individually went down the list and looked at all of those yet.

Q You have never looked at some of these?

A I have. In terms of not necessarily in that order or – or checked off every one as I went down. In other words, I'm still in the process of doing the review is what I'm saying.

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[59] Anthony C. Attard Deposition Transcript, Exhibit 26, page 5.
[60] Exhibit 26, page 10.
[61] Exhibit 26, pages 11-12.

Q Prior to this assignment have you had occasion to look at Dr. Kopp's memo?

A No, not after this, no.

Q What about the Grimes letter?

A No. None o these –

Q The particular ANSI standards that you referred to?

A No. I've heard them thrown around, you know, discussed, but never -[62]

Dr. Attard further testified that he had never worked at a nuclear power plant. His visit to the Millstone Unit 3 spent fuel pool on May 12, 2000 during the discovery proceedings was his first visit to a spent fuel pool.[63]

During the site inspection, Dr. Attard observed a great deal of piping in the spent fuel pool area, including overhead piping. Although Dr. Attard overheard some discussion about the piping involved in the heat removal system above the pool, he testified as follows:

A I was distracted when I was trying to listen. I wanted to listen to the answer that Dr. Thompson asked Mike [Jensen, NNECO employee leading site inspection], so I didn't get to hear the answer about the heat removal system.

Q And what did you learn about the heat removal system?

A. Not very much. I didn't follow up anymore beyond that.[64]

Dr. Attard further testified that he looked for piping directly above the pool because he presumed if there were a leak there it would go straight into the pool. However, Dr. Attard was not concerned that overhead pipe leakage would lead to boron dilution in the pool. As he explained:

[62] Exhibit 26, page 14
[63] Exhibit 26, page 15
[64] Exhibit 26, pages 17-18

> First of all, Mike [Jensen, NNECO's employee] was telling us that – that there are alarms in the control room, if water level rose beyond a certain point, the alarm would go off, or if it drained, it would also alarm.[65]

During discovery, the NRC staff also designated Laurence I. Kopp, Ph.D., as an expert to assist in providing responses to the Intervenors' First Set of Interrogatories and Requests for Production. Inter alia, Dr. Kopp supplied information concerning Intervenors' Interrogatory G-1.

The NRC Staff's response to Interrogatory G-1 ("Please identify all analyses related to the probabilities and consequences of potential criticality incidents and accidents in fuel pools") is as follows:

"None, to the Staff's knowledge. Probability analyses are not part of the spent fuel criticality review process."

At his deposition on May 11, 2000, Dr. Kopp testified that the NRC has no database of incidents involving boron dilution and fuel mishandlings at commercial nuclear reactors.[66] Nevertheless, he conceded the importance of the existence of such a database to proper scientific inquiry.[67] His only knowledge of fuel mishandling incidents at Millstone was based upon three License Event Reports formally filed with the NRC.[68]

Dr. Kopp was asked to comment on evidence that incidents of fuel mishandlings at some nuclear facilities have gone undetected and uncorrected for long periods of time. His comment was: "Well, as I said, the fact that they are eventually found shows that

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[65] Exhibit 26, pages 20-21
[66] Laurence I Kopp Deposition Transcript, Exhibit 27, page 24
[67] Exhibit 27, page 26
[68] Exhibit 27, pages 12-16, 21

eventually the administrative controls did work."[69] Dr. Kopp acknowledged that he had no nuclear power plant operational experience.[70]

When he attended the Unit 3 spent fuel pool site inspection on May 10, 2000, Dr. Kopp observed piping systems above the pool, but he could not identify them, nor was the presence of piping directly above the pool a matter of concern to him. As Dr. Kopp testified:

"I saw pipes and all types of things in there, but I'm not sure what was what. My experience is in criticality and the pool, not in the auxiliary systems, so I wouldn't have any comment on any of that."[71]

During the site inspection, Dr. Kopp also observed that the area suitable for fresh fuel was located only eight inches away from the area not suitable for fresh fuel. After some prodding, Dr. Kopp conceded at deposition that the close proximity of the two regions could be a factor affecting the probability of a fuel misplacement event.[72]

Dr. Kopp was asked to identify the standards applied by the NRC in its review of a spent fuel pool application and what criteria guide the NRC in determining where to draw the line to allow a particular administrative control in a given situation. Dr. Kopp responded as follows:

"Well, for one thing, we would probably look at how many things would have to go wrong for the administrative controls in order to get to a situation where, or example, an erroneous fuel assembly could be put in the wrong place, whether it would take several operators not following a set of preplanned procedures and a second set not verifying the

[69] Exhibit 27, page 36
[70] Exhibit 27, page 40
[71] Exhibit 27, page 43
[72] Exhibit 27, page 43

final position of a fuel assembly. So things like that would be looked at. How many screwups there would have to be to get into an abnormal situation."[73]

Dr. Kopp was unaware of any history of fuel mishandlings at Millstone; he had no knowledge of the incident involving the drop in the Unit 2 spent fuel pool level.[74] However, he acknowledged that such occurrences were pertinent to his review of the instant application.[75] He agreed that an analysis had not been done for multiple misplacement incidents at Millstone, despite a history of actual multiple misplacement incidents at other reactors. He testified that an analysis that would postulate multiple misplacement incidents in the spent fuel pool "would go beyond what the staff requires in reviewing – in spent fuel pool analysis."[76] The following colloquy ensued:

Q And could you point to the standard or the policy that provides that?

A There's one thing, the letter – the letter you referenced before, my letter to – to Mr. Collins, which addressed all the approved methodology and guidance for spent fuel pool criticality analysis, which talks about the single fuel misplacement.

Q Well, where is the standard that establishes that is the extent that will be required by the NRC?

A There is no standard. The – guidance is provided by the double contingency principle.[77]

Finally, Dr. Kopp testified that even if he were aware of a history of Millstone failure to adhere to administrative controls, such information would not inform his analysis of the pending application at all.[78]

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[73] Exhibit 27, pages 45-46
[74] See Robert Griffin Deposition Transcript, Exhibit 25, pages 63-66.
[75] Exhibit 27, pages 47-49
[76] Exhibit 27, page 55
[77] Exhibit 27, page 56
[78] Exhibit 27, pages 64-66

3. Technical Analysis is Insufficient

The technical analysis provided by NNECO is insufficient to support a thorough, independent assessment of the increased burden of risk that would arise from the proposed license amendment. This lack of appropriate analysis provides sufficient grounds for rejection of the license amendment application. Absent such a rejection, the lack of appropriate analysis provides grounds for the conduct of a full evidentiary hearing, at which the missing analysis, or the information that would allow an independent analysis to be performed, could be pursued through cross-examination.

As stated above, NNECO has not provided any estimate of the probability or consequences of a criticality accident, for either present or post-amendment conditions. The absence of such estimates, and of the information that would support such estimates, constitutes one area where the technical analysis provided by NNECO is insufficient.

A second area where technical analysis is insufficient is the lack of full responses to questions A2 and A4 of CCAM/CAM's third set of interrogatories etc to NNECO. (See discussion at pages 28-29.)

Question A4 was intended to obtain an indication of the shape of the envelope of criticality for the pool, as discussed above. NNECO has provided the results of some criticality calculations.[79] However, those results do not establish the envelope of criticality for the pool.

In November 1997, NNECO applied for a revision to the Millstone Unit 3 Technical Specifications. Pursuant to this application, the NRC Staff issued a license amendment in July 1998. These actions followed a determination by NNECO in October 1996 that seismic events could degrade the Boraflex in the fuel pool racks.

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[79] See NNECO's Supplementary Response to Intervenors' Third Set of Interrogatories dated June 21, 2000, Exhibit 28.

In the course of obtaining this license amendment, NNECO examined the integrity of piping in the fuel handling building under seismic loading conditions. As noted above, there are various piping systems in this building that could potentially add water to the pool, causing a dilution of soluble boron.

Consideration of seismic loading does not exhaust the set of scenarios whereby dilution of soluble boron could occur. A complete analysis of boron dilution scenarios requires the consideration not only of seismic loading but of factors that include operation and maintenance errors, heat exchanger tube failures, seal failures, and leakage due to corrosion. Each factor must be considered for each system that could remove water from the fuel pool or add water to the pool. NNECO has not performed a boron dilution analysis that considers each of these factors and systems, and has not provided sufficient information to allow such an analysis to be performed independently.[80]

V. REDUCED REQUIREMENT FOR SURVEILLANCE OF SOLUBLE BORON WILL INCREASE THE PROBABILITY OF A CRITICALITY ACCIDENT

By letter dated April 18, 1999 [sic. 2000], Mr. David Repka, counsel for NNECO, informed the Board of the company's modification to the proposed amendment. NNECO revised its amendment by letter dated April 17, 2000, to require surveillance of the boron concentration in the spent fuel pool every seven days, instead of periodically just during fuel movements.

This proposed revised amendment is consistent with the Standard Technical Specifications and would be acceptable to the Intervenors. However, the NRC Staff

[80] The FSAR does not provide sufficient information to support an independent analysis of boron dilution.

opposed admission of this contention, arguing that it had no merit. Thus, the NRC staff may find no reason to approve the revised amendment as sought by NNECO.

Hence, the Intervenors request that the Board order that no amendment be issued in this proceeding unless it contains a requirement to verify the spent fuel pool's boron concentration at least once every seven days.

The Intervenors' acceptance of the proposed revised amendment does not constitute an acceptance that the presence of soluble boron in pool water can be relied upon as a criticality prevention measure, under either normal or accident conditions. Such reliance is prohibited by GDC 62. Any benefit that soluble boron provides by way of criticality prevention can only be supplemental to a primary and sufficient set of criticality prevention measures that rely on physical systems or processes which do not require support by ongoing administrative controls.

VI. THE PROPOSED LICENSE AMENDMENT FAILS TO COMPLY WITH GDC 62 BECAUSE IT IMPROPERLY RELIES ON ADMINISTRATIVE CONTROLS.

As demonstrated below, the proposed License Amendment Application fails to comply with GDC 62 because it improperly relies on administrative controls for criticality prevention. In addition, the License Amendment Application is inconsistent with the valid and applicable portions of NRC Staff guidance for analysis of criticality prevention measures. CCAM/CAM submits that these issues may be decided as a matter of law, by applying GDC 62 and NRC Staff guidance to the clear and undisputed evidence regarding NNECO's proposed criticality prevention measures. If the Board decides that it is unable to rule for CCAM/CAM on these submissions, the Board should find that CCAM/CAM has raised a genuine, substantial and material factual and legal dispute with NNECO, and order that Contention 6 proceed to a trial pursuant to 10 C.F.R. § 2.1115.

The Licensing Board must allocate the burden of proof to the Applicant in considering whether the standard for going forward with an adjudicatory hearing is satisfied.

A. The General Design Criteria Establish Minimum Design Requirements for Nuclear Power Plants.

The Commission's General Design Criteria ("GDC") for Nuclear Power Plants establish the basic principles of nuclear power plant design. They constitute:

> *minimum requirements* for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the [Nuclear Regulatory] Commission.

Appendix A to 10 C.F.R. Part 50, Introduction (emphasis added). The General Design Criteria constitute basic guidance for the more detailed NRC safety regulations. They are "intended to provide engineering goals rather than precise tests or methodologies by which reactor safety [can] be fully and satisfactorily gauged." *Petition for Emergency and Remedial Action*, CLI-78-6, 7 NRC 400, 406 (1978), *quoting Nader v. Nuclear Regulatory Commission*, 513 F.2d 1045 (D.C. Cir. 1975). As the Commission noted in that case, there are a "variety of methods for demonstrating compliance with GDC," including regulatory guides, standard format and content guides for license applications, the Standard Review Plan, and Branch Technical Positions. *Id.*

Although the Commission allows flexibility in developing methods for compliance with the general requirements of the General Design Criteria, the fundamental principles of the GDC must be adhered to in choosing those methods. Thus, for example, in *Nader v. Ray*, the Court of Appeals held that a set of detailed standards for prevention of a loss of coolant accident was consistent with the broad requirement of GDC 35 for a "system to provide abundant emergency core cooling." 513 F.2d at 1051-

53. *But see Consumers Power Co.* (Big Rock Point Nuclear Plant), ALAB-725, 17 NRC 562, 567 571 (1983).[81]

B. The Plain Language of GDC 62 Requires the Use of Physical Systems or Processes to Prevent Criticality, and Thereby Precludes the Use of Administrative Controls.

1. The plain language of GDC 62 requires the use of physical systems or processes to prevent criticality.

General Design Criterion 62 is entitled "Prevention of criticality in fuel storage and handling." GDC 62 instructs that:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by the use of geometrically safe configurations.

The language of GDC 62 is quite clear: criticality control measures must be carried out by physical systems or processes. The phrase "physical systems or processes" is not

[81] In *Consumers Power*, the Appeal Board found that a remotely controlled makeup line for the spent fuel pool constituted a "physical system" for criticality control, and therefore was consistent with the requirement of GDC 62 that criticality must be maintained through "physical systems or processes." *Id.* at 571. In CCAM/CAM's view, the use of a makeup line is an impermissible administrative procedure, because it requires ongoing reliance on human action to turn on the flow of water into the makeup line. Two aspects of the *Consumers Power* decision give it questionable applicability to this case, however. First, the Appeal Board noted that it had been provided with "no evidence" to suggest that the make-up line was not a physical system within the "broad, general terms" of the GDC. 17 NRC at 571. Here, in contrast, CCAM/CAM has provided the Board with evidence of (a) the clear basis for distinguishing physical measures from ongoing administrative controls, and (b) the Commission's intent to preclude the use of procedural controls for criticality control. *See* Sections B.1.a and B.1.b, below. Second, the circumstance addressed in the *Consumers Power* decision, involving the hypothetical exposure of high-reactivity (fresh or nearly-fresh) fuel to boiling water, foam or mist, is now implicitly addressed in Staff guidance which establishes a Keffective value of 0.98 for such a scenario, rather than requiring measures for maintaining Keffective below 0.95. *See* Kopp Memorandum at 4-5 (Exhibit 4). The Staff guidance is provided in the context of fresh fuel storage in a new fuel storage facility (vault), but logically must apply to pool storage of high-reactivity fuel that could become critical in the presence of boiling water, foam or mist. Indeed, the informational Appendix A to ANSI/ANS-8-17-1984, American National Standard, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors (January 13, 1984), indicates that "void formation by boiling" is a normal condition for the purpose of evaluating the potential for criticality in a fuel pool. Thus, the question of whether a makeup line

defined in Appendix A to Part 50, but it may be understood by reference to the example provided in GDC 62 of an acceptable physical system or process: a geometrically safe configuration. In other words, fuel storage racks must be configured in such a way as to prevent criticality, without resort to any ongoing administrative controls. Standing alone, the plain language of GDC 62 clearly dictates that NNECO must rely solely on physical measures to avoid criticality. Because NNECO intends to rely in part on ongoing administrative controls, *i.e.*, control of burn-up, fuel age and enrichment, its license amendment application must be rejected based on the plain language of GDC 62.

Moreover, in contrast to some of the other General Design Criteria, nothing about GDC 62 remains open-ended or subject to later revision. For instance, with respect to the definition of a loss of coolant accident, footnote 1 of Appendix A to Part 50 states that "[f]urther details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development." Thus, GDC 62 is distinct from other criteria that "have not as yet been suitably defined." *Nader v. NRC,* 513 F.2d at 1052.

2. Physical systems and processes are distinct in nature from ongoing administrative controls

In the prehearing conference, members of the Licensing Board questioned the distinction between physical systems and processes and administrative controls. Concededly, any physical measure has some administrative component, and any administrative measure has a physical component. However, there is a basic difference between the nature of physical systems and processes, on the one hand, and administrative controls, on the other hand.

If a subcritical margin of reactivity is to be maintained in a fuel pool solely by use of a geometrically safe configuration, then administrative controls will be needed to

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constitutes a physical measure for purposes of eliminating a boiling, misting or foam environment in a spent fuel pool has effectively been mooted.

ensure that the fuel racks provide the required configuration. That configuration must be maintained during normal operation and after specified insults, such as an earthquake or the drop of an object onto a rack. The necessary administrative controls may be stringent, but they will be applied on a one-time basis. After the fuel racks are designed, fabricated and installed, ongoing administrative controls will not be required.

Similarly, if a subcritical margin of reactivity is to be maintained in a pool partly by exploiting the neutron-absorbing properties of the fuel racks, then one-time administrative controls will be needed to ensure that those properties are provided. For example, if Boral panels are attached to the racks, then one-time administrative controls will be needed to ensure that the Boral panels are properly designed, fabricated and installed. Periodic inspections may be needed to ensure that the Boral panels or other neutron-absorbing materials retain their needed properties, but these inspections will be comparatively straightforward.

By contrast, prevention of criticality by ongoing administrative controls will require continuing actions by human beings to carry out these measures, such as inputting information into a computer system, and operating and maintaining equipment. These measures must be carried out throughout the period when criticality is possible. For example, if the presence of soluble boron is to be exploited as a means of criticality suppression in a fuel pool, then administrative controls must ensure that the concentration of soluble boron in the pool water never falls below a specified level. These administrative controls must be implemented on a continuous, ongoing basis, with complete reliability. The controls must apply to an entire pool, and to canals or other pools that are interconnected with that pool.

Similarly, if restrictions on fuel burnup/enrichment or fuel age are to be exploited as means of criticality suppression in a rack in a fuel pool, then ongoing administrative controls must ensure that a fuel assembly is never placed in the rack unless its burnup/enrichment or age is within a specified range. Ongoing administrative controls on fuel burnup/enrichment or fuel age can be specified for an entire pool, for a particular rack, or for particular spaces within a rack. At a number of nuclear plants, a "checkerboard" pattern of fuel placement has been specified, wherein particular spaces in the repeating checkerboard pattern have particular restrictions on fuel burnup/enrichment. These administrative controls must be effective on each occasion when a fuel assembly could be placed in the pool.

Ongoing administrative controls are inherently less reliable than physical systems and processes, because they involve the repetition of tasks numerous times, thus providing multiple and cumulative opportunities for error. They must also be implemented by human beings, and thus are prey to human error. A related factor noted by the NRC Staff in an Information Notice is the potential unfamiliarity of fuel handling personnel with procedures:

> Refueling activities are safety-significant operations that are not conducted on a routine basis. In addition, fuel handling activities are often performed by contractor personnel under the supervision of licensee personnel. As a result, fuel handling personnel may not be familiar with the fuel handling equipment or may feel that their experience in fuel handling operations permits them to ignore some requirements for procedural use and adherence.

Information Notice 94-13 (February 22, 1994).[82]

Thus, while physical systems and processes entail some administrative controls, these are one-time controls that generally are completed before the system or process is put to use. By contrast, the use of restrictions on fuel burnup/enrichment or fuel age, or reliance on the presence of soluble boron, as means of criticality suppression will require ongoing administrative controls. This requirement can never be relaxed, and the controls must be implemented on a completely reliable basis. Over time, ongoing administrative controls of this kind will have a much higher cumulative probability of failure than one-time controls.

82 A copy of this Information Notice is attached as Exhibit 29.

C. The Rulemaking History of GDC 62 Supports the Plain Language of the Regulation.

The rulemaking history of GDC 62 makes it even more clear that in promulgating GDC 62, the Commission intended to impose the fundamental requirement that criticality must be controlled by physical rather than administrative or procedural measures. Early in the rulemaking process, and in the proposed rule, the Commission considered language favoring physical systems or processes, but permitting procedural measures. In response to comments, however, the Commission removed the reference to procedural measures, and established a clear requirement that physical systems and processes must be used. In addition, while the General Design Criteria were originally proposed as guidance, they ultimately were promulgated in the form of minimum requirements.

1. Pre-rulemaking documents

To CCAM/CAM's knowledge, a set of draft General Design Criteria first appeared as an attachment to an Atomic Energy Commission ("AEC")[83] press release of November 22, 1965, entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits."[84] The attachment included draft Criterion 25, which proposed the following language relating to prevention of criticality in fuel handling and storage facilities:

> The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

During the following year, the AEC continued to revise the language of the proposed GDC in response to comments made by AEC staff and by members of the

83 The Atomic Energy Commission was the predecessor agency to the NRC.

84 The Press Release and attached documents are attached as Exhibit 30.

Advisory Committee on Reactor Safeguards ("ACRS"). A revised draft of October 6, 1967, prepared by the AEC, contained draft Criterion 10, which stated:

> Possibilities for inadvertent criticality must be prevented by engineered systems or processes to every extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.[85]

The same language appeared again in an October 20, 1966 draft, which was attached to a letter of October 25, 1966 from J.J. DiNunno of the AEC to David Okrent of the ACRS.[86]

Another draft of a GDC for criticality prevention appears as a February 6, 1967, attachment to a letter from J. J. DiNunno of the AEC to Nunzio J Palladino of the ACRS, dated February 8, 1967.[87] In this draft, the potential for criticality in fuel handling and storage facilities was addressed by Criterion 61, which stated:

> Possibilities for criticality in new and spent fuel storage shall be prevented by physical systems or processes to every extent practicable. Such means as favorable geometries shall be emphasized over procedural controls.

2. Proposed GDC for criticality control

On June 16, 1967, the AEC Director of Regulation proposed a set of draft GDCs to the AEC Commissioners, "for consideration by the Commission at an early date".[88] The set of GDCs was described as a proposed amendment to 10 CFR 50. The potential

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85 Internal AEC memorandum from G.A. Arlotto to J.J. DiNuuno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966), attached as Exhibit 31.

86 The October 25, 1966, letter and attached draft are attached to this Summary as Exhibit 32.

87 The February 8, 1967 letter and attached draft are attached to this Summary as Exhibit 33.

88 Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967). The Note and relevant excerpts from Appendix B to the Note are attached as Exhibit 34.

for criticality in fuel handling and storage facilities was addressed by draft Criterion 66, which stated:

> Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Shortly thereafter, this language appeared in the Commission's notice of proposed rulemaking for the General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967).[89] Thus, throughout the early development of the GDC for criticality control, the concept of procedural controls was included in the language of the criterion.

The introduction to the General Design Criteria stated that they were "intended to be used as guidance in establishing the principal design criteria for a nuclear power plant." 32 Fed. Reg. at 10,215.

3. Comments on the proposed rule

Comments on the proposed GDC show persistent effort by the nuclear industry to influence the evolution of many of the GDCs, but comparatively little concern about the criterion that became GDC 62. The Commission did, however, receive an influential comment on criticality prevention from the Nuclear Safety Information Center, Oak Ridge National Laboratory (ORNL).[90] The ORNL commented as follows:

> We do not understand the implication of 'or processes' at the end of the first sentence, nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: 'Such means

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89 A copy of the Federal Register notice is attached to this Summary as Exhibit 35.

as geometrically safe configurations shall be used to insure that criticality cannot occur.'[91]

On July 15, 1969, the AEC prepared a set of revisions to the GDC, based on comments by the ACRS and the nuclear industry. As discussed in the accompanying cover letter, a major difference between the proposed GDC and the revised GDC was that the revised GDC "[e]stablish "minimum requirements" for water-cooled reactors, whereas the published criteria were "guidance" for all reactors.[92] The revised GDC included GDC 62, entitled "Prevention of Criticality in Fuel Storage and Handling:"

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

On June 4, 1970, the AEC prepared another revision to the GDC, containing the identical language of GDC 62 that had been prepared on July 15, 1969. This revision was circulated to other members of the AEC and the Atomic Industrial Forum (AIF), a nuclear industry trade organization.[93] Although the AIF recommended substantial changes to other GDCs contained in the revised draft, it accepted the new draft GDC 62 without any proposed alteration.

4. The Final Rule

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[90] ORNL's comments on the proposed rule were contained in an attachment to a letter of September 6, 1967 from William B. Cottrell of ORNL to H. L. Price of the AEC, attached as Exhibit 36.

[91] *Id.*, Attachment containing "Specific Comments" at 11.

[92] Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Units (July 15, 1969), attached as Exhibit 37.

[93] *See* Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970) Attached to the Wiggin letter is a marked-up version of the June 4, 1966, revised draft of the GDC. The Case Memorandum and enclosed documents are attached as Exhibit 38.

On February 20, 1971, the AEC published the General Design Criteria in final form.[94] The introduction to the GDC's now characterized them as "minimum requirements" for the design of nuclear power plants, rather than "guidance" as had been proposed. In addition, the final rule included GDC 62, which provided that:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

The final rule removed the language in the proposed rule that had included "procedural controls" in the set of acceptable measures for controlling criticality. Instead, "physical systems or processes" became the only acceptable means of criticality control. Moreover, geometrically safe configurations were clearly identified as the "preferred" type of physical system or process, in lieu of "emphasized" controls. It can be assumed that ORNL's comment regarding the impracticality of procedural controls had an important influence on this near-final step in the evolution of GDC 62. Thus, the rulemaking history of GDC 62 illustrates the importance placed by the Commission on physical systems and processes, in contrast to procedural controls.

D. The Plain Language of GDC 62 Is Not Altered or Contradicted By Other Relevant NRC Criticality Standards.

GDC 62's plain language, requiring the use of physical systems or processes to prevent criticality, is consistent with other relevant NRC regulations for criticality prevention that were promulgated afterwards. In particular, GDC 62 is consistent with the NRC's requirements for criticality prevention in 10 C.F.R. § 50.68 and 10 C.F.R. § 70.24. Both the language of these regulations and their regulatory history demonstrate that the Commission considers physical systems and processes to be essential to preventing criticality in the storage of spent or fresh fuel.

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94 Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971). A copy of the Federal Register notice is attached as Exhibit 39.

1. 10 C.F.R. §§ 70.24 and 50.68

Aside from GDC 62, prior to 1998 the NRC's only criticality-related regulation for operating nuclear power plants consisted of 10 C.F.R. § 70.24, which required criticality monitoring for any licensee authorized to possess significant quantities of special nuclear material ("SNM"). The regulation included a provision authorizing licensees to seek an exemption where good cause was shown. 10 C.F.R. § 70.24(d).

On December 3, 1997, the NRC concurrently published in the Federal Register a proposed rule and a direct final rule, making changes to 10 C.F.R. § 70.24 and adding a new section 50.68.[95] The purpose of the amended regulations was to eliminate the requirement for case-by-case exemptions from § 50.24, and establish a blanket exemption for licensees who agreed to follow a set of criticality accident prevention requirements in the new section 50.68. The new set of rules was based on the NRC's experience that a "large number of exemption requests ha[d] been submitted by power reactor licensees and approved by the NRC based on safety assessments which concluded that the likelihood of criticality was negligible."[96] The discussion of safety in criticality control which followed this assertion made it clear that the finding of negligible risk was based in part on the assumption that during fuel storage, physical measures such as design features would be used to prevent criticality:

> At a commercial nuclear power plant, the reactor core, the fresh fuel delivery area, the fresh fuel storage area, the spent fuel pool, and the transit areas among these, are areas where amounts of SNM sufficient to cause a criticality exist. In addition, SNM may be found in laboratory and storage locations of these plants, but an inadvertent criticality is not considered credible in these areas due to the amount and configuration of the SNM. The SNM that could be assembled into a critical mass at a commercial nuclear power plant is only in the form of nuclear fuel. Nuclear power plant licensees have procedures and the plants have design

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[95] Proposed Rule, Criticality Accident Requirements, 62 Fed. Reg. 63,911; Direct Final Rule With Opportunity to Comment, Criticality Accident Requirements, 62 Fed. Reg. 63,825.

[96] 62 Fed. Reg. at 63,825, Col. 3.

features to prevent inadvertent criticality. The inadvertent criticality that 10 CFR 70.24 is intended to address could only occur during fuel-handling operations.

In contrast, at fuel fabrication facilities SNM is found and handled routinely in various configurations in addition to fuel. Although the handling of SNM at these facilities is controlled by procedures, the variety of forms of SNM and the frequency with which it is handled provides greater opportunity for an inadvertent criticality than at a nuclear power reactor.

At power reactor facilities with uranium fuel nominally enriched to no greater than five (5.0) percent by weight, the SNM in the fuel assemblies cannot go critical without both a critical configuration and the presence of a moderator. *Further, the fresh fuel storage array and the spent fuel pool are in most cases designed to prevent inadvertent criticality, even in the presence of an optimal density of unborated moderator.* Inadvertent criticality during fuel handling is precluded by limitations on the number of fuel assemblies permitted out of storage at the same time. *In addition, General Design Criterion (GDC) 62 in Appendix A to 10 CFR Part 50 reinforces the prevention of criticality in fuel storage and handling through physical systems, processes, and safe geometrical configuration.* Moreover, fuel handling at power reactor facilities occurs only under strict procedural control. Therefore, the NRC considers a fuel-handling accidental criticality at a commercial nuclear plant to be extremely unlikely. The NRC believes the criticality monitoring requirements of 10 CFR 70.24 are unnecessary *as long as design and administrative controls are maintained.*[97]

Thus, in promulgating § 50.68, the Commission affirmed the language of GDC 62 which restricts criticality prevention measures to physical systems and processes.

The language of § 50.68, as it was finally promulgated, contains a list of measures for criticality prevention that can be implemented in lieu of maintaining a criticality monitoring system.[98] Although these provisions contain some references to procedures and administrative measures, they do not undermine or contradict the general requirement of GDC 62 for physical criticality prevention measures. For instance, subsection (b)(1) requires that:

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

This provision simply requires licensees to have a procedure which forbids them from handling or storing any fuel assemblies for which the licensees are unable to maintain subcriticality. It does not explicitly address whether, for the number of assemblies that *are* permitted to be handled or stored, criticality control must be accomplished through physical measures or may be addressed by administrative measures. However, it is noteworthy that the provision assumes that at least one administrative measure, reliance on the presence of boron in the pool water, will not be available.

Subsections (b)(2) and (b)(3) provide that:

> (2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage rack shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

> (3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly

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97 62 Fed. Reg. at 63,825-26. (emphasis added)

98 *See* Final Rule, Criticality Accident Requirements, 63 Fed. Reg. 63,127 (November 12, 1998).

reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

These requirements relate to the storage of fresh fuel in fresh fuel storage racks. Fresh fuel storage racks are free-standing racks that surround the fresh fuel with air. By design, no water is present that could act as a moderator. The absence of water as a moderator is a physical system or process for criticality control, built into the design of the fresh fuel storage facility. This is consistent with GDC 62.

Subsections (b)(2) and (b)(3) require the licensee to perform an accident analysis that demonstrates criticality will be prevented, even if water accidentally enters the fresh fuel racks. A licensee may be exempted from the accident analysis if it demonstrates one of two things: that flooding will be prevented by administrative measures, or that fresh fuel storage racks will not be used. The first option, use of administrative measures to prevent flooding, is *in addition to* the design features by which fresh fuel racks are located in a place that is removed from the presence of water. Thus, it cannot be viewed as a primary criticality prevention measure, but as a secondary measures used as a back-up to the primary design features. If the second option is elected, the licensee must show that fresh fuel racks are not used, *i.e.*, that the fresh fuel is stored in a fuel pool. If fresh fuel is stored in a pool, it must meet the same criticality prevention requirements as apply to spent fuel (*see* subsection (b)(4), discussed below). Under these requirements, the fuel must remain subcritical, even in the absence of soluble boron.[99] Accordingly, there is nothing about subsections (b)(2) or (b)(3) that is inconsistent with the requirement of GDC 62 that physical systems and processes must be used to prevent criticality.

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99 As discussed in note 81 above, arrangements for storage of fresh fuel in a pool should also ensure that the fuel remains subcritical in the presence of boiling water, foam or mist.

Subsection (b)(4) relates to the storage of fuel in spent fuel pools. Although this provision also mentions administrative measures in the sense that it discusses the parameters for taking credit for the presence of soluble boron in the water, the provision also makes it clear that criticality ultimately must be prevented *without* resort to administrative controls:

> If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

Thus, the basic requirement of subsection (b)(4) is that criticality must be controlled (*i.e.*, Keffective maintained below 0.95 or 1.0, depending on the taking of credit for soluble boron) without considering the presence of soluble boron in the water.[100] Moreover, this control must be achieved for "racks loaded with fuel of the maximum fuel assembly reactivity".

The Board asks parties to this proceeding to define the term "maximum fuel assembly reactivity."[101] For the Millstone Unit 3 fuel pool, "fuel of the maximum fuel assembly reactivity" is fresh fuel with an enrichment of 5%, because that is the most reactive fuel that will pass through the pool. Moreover, NNECO's application does not seek credit for soluble boron. Thus, subsection (b)(4) establishes a requirement that Keffective in the Millstone Unit 3 pool must not exceed 0.95, at a 95 percent probability,

[100] The other provisions of § 50.68, subsections (b)(5) through (8), are not relevant to this proceeding.
[101] ASLB Memorandum (Questions for Parties), May 23, 2000.

95 percent confidence level, if the pool is flooded with unborated water and the racks are loaded with fresh fuel with an enrichment of 5%.

NNECO has provided the results of criticality calculations which show that this requirement would be violated for a full loading of fresh fuel in each of the proposed three regions of the pool.[102] In Region 1, Keffective would be 0.9728 with no soluble boron. In Region 2, Keffective would be 0.9842 for a boron concentration of 2,000 ppm (and would be higher at lower boron concentrations). In Region 3, Keffective would be 0.9811 for a boron concentration of 1,320 ppm (and would be higher at lower boron concentrations).

Thus, NNECO has shown that its proposed license amendment would violate subsection (b)(4) of section 50.68. That violation provides sufficient grounds for denial of NNECO's application.

It should also be noted that the type of ongoing administrative measure proposed by NNECO in the instant case, *i.e.*, control of burnup/enrichment or fuel age, is not condoned by § 50.68, or even mentioned.

2. 10 C.F.R. § 72.124

The Commission has also promulgated regulations for control of criticality at Independent Spent Fuel Storage Installations ("ISFSI's"). These regulations are inconsistent with GDC 62, because they do not unequivocally require the use of physical systems or processes for criticality control, and instead apply a practicability standard. 10 C.F.R. § 72.124(b) provides as follows:

> *Methods of criticality control.* When practicable the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing

[102] Exhibit 28.

materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy.

The ISFSI regulations do not apply to the instant proceeding, however. The Harris operating license amendment is being considered under Part 50 of the regulations, which govern nuclear power

Section 72.124(b) is also inapplicable to this case because design and operation of an ISFSI is fundamentally different than the design and operation of a nuclear power plant, such that the Commission might have grounds for establishing a more relaxed standard for criticality control at ISFSI's than for nuclear power plants. As recognized by the Commission in the preamble to the ISFSI regulations, an ISFSI is "not coupled to either a nuclear power plant or a fuel reprocessing plant." 43 Fed. Reg. at 46,309. The Commission saw "a need for a new regulation covering the requirements for extended spent fuel storage under *static storage conditions involving no operations on such materials*." *Id.* (emphasis added). In contrast, the operations in a fuel storage building of a nuclear power plant cannot be considered "static." Fresh fuel is constantly being brought into the fuel building and moved through the fuel transfer canals and pools into the reactor. The same equipment and personnel are used to move both fresh and spent fuel. Also, at a nuclear power plant there will be occasions when spent fuel with a reactivity nearly as high as, or even higher than, the reactivity of fresh fuel is stored in fuel pools. This could occur, for example, during a full core offload.

Thus, at an operating nuclear power plant there is the constant possibility that fresh fuel will be placed inappropriately into a spent fuel storage pool. Indeed, such mispositioning has occurred in the past.[103] By requiring physical systems and processes for the control of criticality, GDC 62 ensures that criticality will be avoided, regardless of the burnup level or age of fuel that is placed in the pool. It is much less likely that fresh or highly reactive fuel would be placed in an ISFSI, and thus there may not be the same need to insist on physical measures for criticality prevention at an ISFSI.

Although the Board need not reach this far in finding that 10 C.F.R. § 72.142(b) has no precedential value in this case, it is also noteworthy that § 72.142(b) was not duly promulgated in compliance with the procedural requirements of the Administrative Procedures Act, 5 U.S.C. § 553, for public notice and opportunity to comment. The current language of § 72.124(b) was promulgated in 1988, when the Commission added requirements for Monitored Retrievable Storage ("MRS") to the ISFSI regulations.[104] The 1988 rulemaking fundamentally altered the Commission's existing regulation for criticality control at ISFSI's, which had been promulgated with the original set of ISFSI regulations in 1980.

Section 72.73(b) of the original ISFSI regulations explicitly and unequivocally required the use of geometric spacing and/or fixed neutron-absorbing material – *i.e.*, physical systems and processes – for criticality control:

> *Methods of criticality control.* The design of an ISFSI or MRS must be based on favorable geometry (spacing), permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy. In criticality design analyses for underwater storage systems, credit can be taken for the neutron absorption of rack structures and the water within the storage unit.

Final rule, Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation, 45 Fed. Reg. 74,693, 74,710 (November 12, 1980).

On May 27, 1986, the Commission proposed to amend the Part 72 regulations to encompass the licensing of MRS facilities and to "clarify matters that have arisen since part 72 was made effective on 11/28/80." Proposed Rule, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, 51

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[103] *See* examples cited in Appendix B.

Fed. Reg. 19,106. The Federal Register notice included the following provision for methods of criticality control, § 72.93:

> *Methods of criticality control.* The design of an ISFSI or MRS must be based on favorable geometry (spacing), permanently fixed neutron absorbing materials (poisons), or both. In criticality design analyses, credit can be taken for fixed neutron absorbing material present within the storage structure.

51 Fed. Reg. at 19,124. These proposed changes to the 1980 criticality control regulation were minor: they added a reference to an MRS, and they took out the sentence requiring the verification of continued efficacy of fixed poisons. Significantly, the proposed rule continued to require the use of favorable geometry and permanently fixed poisons as mandatory measures.

When the final rule was promulgated in 1988, the provision governing methods for controlling criticality was transformed. No longer did the rule contain a mandatory requirement for favorable geometry and fixed poisons; instead, these measures were called for only "if practicable." The Commission had also added to § 72.124(a) the following "double contingency" provision, not found in the 1980 rule or the 1986 proposed rule:

> Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.[105]

[104] Final Rule, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, 53 Fed. Reg. 31,651 (August 19, 1988).

[105] 53 Fed. Reg. at 31,674. The 1980 rule and the proposed 1986 rule had provided that: Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to prevent a nuclear criticality accident. 45 Fed. Reg. at 74,710; 51 Fed. Reg. at 19,124.

No justification can be found in the preamble to the final rule for this eleventh hour substitution of language that was so completely different from the proposed rule. The only mention of the changes is the following discussion:

> *Comment:* A comment was received concerning the removal of the requirement for verifying continued efficacy of solid neutron poisons.
> *Response:* Several changes have been made to the criticality section of the final rule to make it correspond to other Parts of the Commission's regulations and standard criticality review practices. Verification of solid neutron poisons has been retained. Double contingency criteria and requirements for criticality monitors have been added. It is not the intent of the revision concerning criticality monitors to require monitors in the open areas where loaded casks are positioned for storage as that system is static. Monitors are required where the systems are dynamic.

53 Fed. Reg. at 31,656. Here, the Commission effectively admitted that the changes had nothing to do with a response to comments: the provision relating to the comment regarding verification of the continued efficacy of solid neutron poisons was not changed at all, but was "retained." Instead, the Commission claimed to have changed the rule "to make it correspond to other Parts of the Commission's regulations and standard criticality review practice." The Commission did not identify what other regulations this new rule is consistent with, and indeed none can be identified: this is a rationalization without substance. Nor did the Commission attempt to describe the alleged "standard criticality review practice," justify it, or explain why the Commission failed to give public notice prior to making the change. By making such a major substantive change in the final rule, without first providing public notice or permitting public comment, the Commission violated the Administrative Procedure Act, which renders the rule invalid.[106]

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[106] *See American Frozen Food Institute v. Train*, 539 F.2d 107, 135 (D.C. Cir. 1976); *Connecticut Light and Power Co. v. NRC*, 673 F.2d 525, 533 (D.C. Cir.), *cert. denied*, 459 U.S. 835 (1982); *Florida Power & Light Co. v. U.S.*, 846 F.2d 765, 771-72 (D.C. Cir. 1988), *cert. denied*, 490 U.S. 1045 (1989); *Air Transport Association of America v. FAA,* 169 F.3d 1, 6-8 (D.C. Cir. 1999).

E. The Administrative Criticality Prevention Proposed by NNECO Would Violate GDC 62.

As described above in Section III.B, CP&L proposes to employ ongoing administrative controls on the burnup/enrichment and age of fuel, in order to suppress criticality under normal conditions.

This reliance on ongoing administrative controls violates the language and intent of GDC 62, which is to ensure that *physical systems and processes*, preferably geometrically safe configuration of the assemblies, are used to control criticality. Similarly, NNECO relies on the presence of soluble boron to prevent criticality under accident conditions. This violates the plain meaning and intent of GDC 62, because the introduction and maintenance of soluble boron in the spent fuel pools require ongoing administrative actions and procedures, and do not constitute physical systems or processes.[107]

F. NNECO's Proposed Reliance on Administrative Criticality Prevention Measures Is Not Justified by Draft Reg. Guide 1.13 or Other NRC Staff Guidance.

In opposing the admissibility of Contention 6, NNECO and the NRC Staff argued that reliance on control of burnup/enrichment levels and fuel age to prevent criticality is permitted by Draft Reg. Guide 1.13. The Commission has stated generally that "if there is conformance with regulatory guides, there is likely to be compliance with the GDC." *Petition for Emergency and Remedial Action,* CLI-78-6, 7 NRC 400, 406 (1978). As a Board in a related case has recognized, however, this is "not a blanket endorsement of the notion that regulatory guides necessarily govern." LBP-99-25, 50 NRC at 35. Where there is inconsistency between a regulation and a regulatory guide, the regulation is controlling. A regulation has the force of law; in comparison, a regulatory guide is a set

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[107] In one criticality analysis, NNECO relies on the presence of soluble boron during an accident. *See* NNECO's June 21, 2000, Supplementary Response (Exhibit 26). This response shows that 30 ppm of soluble boron would be required to prevent criticality (Keffective = 1.0) if one fresh fuel assembly is mis-loaded into a Region 3 rack and the pool temperature is 150 degrees F.

of recommendations setting forth acceptable methods for complying with the regulation. Such documents "are useful as guides," but "insofar as the adjudicatory process is concerned, they represent the opinions of one of the parties to that process and as such cannot be viewed as necessarily controlling." *Potomac Electric Power Co.* (Douglas Point Nuclear Generating Station, Units 1 and 2), LBP-76-13, 3 NRC 425, 432 (1976). *See also Louisiana Energy Services* (Claiborne Enrichment Center), LBP-91-41, 34 NRC 332, 354 (1991). Therefore, a Reg. Guide cannot be relied on to modify or circumvent the requirements of duly promulgated regulations like the General Design Criteria.

To the extent that they permit prevention of criticality through administrative procedures and controls, Draft Reg. Guide 1.13 and the Kopp Memorandum violate the plain language and intent of GDC 62. Therefore, in this respect they must be disregarded.

G. Neither NNECO Nor the Staff Has Demonstrated That Public Health And Safety Will Be Adequately Protected If NNECO Relies on Ongoing Administrative Controls for Criticality Prevention.

Although the Staff's regulatory guidance is fundamentally at odds with GDC 62, the Staff's practice of permitting ongoing administrative controls for the prevention of criticality in spent fuel pools is well-entrenched. In recent years, the NRC Staff has approved many applications similar to NNECO's, setting a trend toward higher and higher density of spent fuel storage and greater and greater reliance on administrative controls to prevent criticality.

Astoundingly, the Staff has pursued this course for over two decades without conducting any safety analysis to determine whether its radical departure from the requirements of GDC 62 could be justified on safety grounds. The Staff has never done a systematic analysis of the potential for criticality accidents when reliance is placed on administrative measures instead of physical measures. Although the Staff has advocated the Double Contingency Principle in evaluating criticality accidents since 1978, it has made no attempt to determine what combinations of fuel handling or pool management errors would violate the Double Contingency Principle. Instead, as discussed above and in Appendix A, it has merely watered down the Double Contingency Principle to a Single Contingency Principle. Despite the many years of accumulated licensee experience with spent and fresh fuel storage, the Staff has never attempted to conduct a systematic review of the operating experience of licensees with fuel mispositioning or fuel incidents relevant to boron dilution.[108] The Staff does not even maintain a systematic data base of the experience of nuclear power plant licensees with such problems as mispositioning of fuel assemblies and soluble boron management errors.

In fact, as shown in Appendix B, available (but incomplete) information shows that there is a significant history of incidents relevant to failure of criticality prevention in fuel pools. These incidents include mispositioning of fuel assemblies and incidents relevant to boron dilution, including one boron dilution event. Significantly, the record includes events in which a single error resulted in the mispositioning of more than one fuel assembly, such as the mispositioning of 184 fresh fuel assemblies in the Oyster Creek spent fuel pool in 1986. The record also includes incidents that are relevant to the prevention of criticality solely through the use of physical systems and processes, notably some errors in criticality analyses. These incidents raise questions about the size of the safety margin achieved when preventing criticality solely through the use of physical systems and processes, and the wisdom of cutting into that safety margin by placing reliance on less-reliable ongoing administrative measures.

As set forth in Appendix C, experience at U.S. nuclear power plants shows that fuel mispositioning, involving placement in a pool of one or more fuel assemblies with inappropriate burnup/enrichment or age, is a likely occurrence. Experience also shows that the concentration of soluble boron in a pool can fall below specified levels. Some accident sequences could yield substantial reductions in soluble boron concentration. From a qualitative perspective, it is clear that criticality scenarios which involve the

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108 CCAM/CAM is aware of only one generic study of boron dilution, which was done by a self-interested party, the Westinghouse Corporation, and which failed to summarize the historical record of

failure of ongoing administrative controls have a much higher probability of occurring than criticality scenarios involving failure of physical controls. Also, Appendix C shows that significant onsite and offsite radiation exposures are potential outcomes of a criticality event in a fuel pool. Under the circumstances, there is no basis for concluding that the public health and safety can be protected through reliance on administrative measures for criticality prevention at Millstone Unit 3.

H. NNECO's Criticality Accident Analysis Misapplies Applicable Staff Guidance.

As discussed above, NNECO's criticality analysis is fundamentally deficient because NNECO relies on ongoing administrative controls for criticality prevention, in violation of GDC 62. To the extent that it condones this unlawful practice, current NRC guidance is also invalid.

In examining the lawfulness and reasonableness of NNECO's criticality prevention measures, it is necessary to go beyond a determination that physical systems and processes are required for criticality prevention. Even where such physical measures are used and are effective in preventing criticality during normal operation, it is necessary to perform an accident analysis to determine whether such measures are adequate to prevent criticality under a range of accident conditions. For this purpose, portions of the NRC Staff's guidance for criticality control provide useful guidance that is consistent with GDC 62. In particular, the Double Contingency Principle provides a method of analysis that is useful for evaluating the potential for criticality accidents.

As set forth in Draft Reg. Guide 1.13, the Double Contingency Principle requires a nuclear criticality safety analysis to demonstrate that criticality could not occur "without at least two unlikely, independent, and concurrent failures or operating limit violations." NNECO has misapplied this guidance in five principal respects. First, NNECO ignores the words "at least," and evaluates only one failure instead of sets of

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relevant events. *See* Appendix C.

failures; second, it fails to determine what failures are "unlikely, independent, and concurrent;" third, it assumes that mispositioning of fuel is an "unlikely" event when in fact it is likely; fourth, it unreasonably assumes that a single error can lead to the mispositioning of only one fuel assembly; and fifth, it considers failures of criticality prevention measures that are forbidden by GDC 62.

VII. CONCLUSION

For the foregoing reasons, the License Amendment Application for the expansion of the Millstone Unit 3 spent fuel storage capacity must be rejected. As a matter of law, the criticality prevention measures proposed are inconsistent with GDC 62 and valid and applicable NRC Staff guidance. NNECO's criticality prevention measures are demonstrably insufficient to provide a reasonable level of protection to public health and safety; NNECO's proposed reliance on new and complex administrative controls pose an undue and unnecessary risk of a criticality accident.

If the Board declines to reject the application as a matter of law, it should find that the Intervenors have raised material and substantial issues of law and fact and order the parties to proceed to an adjudicatory hearing on their contentions.

Respectfully submitted,

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