UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-423-LA-3
NORTHEAST NUCLEAR ENERGY	)	
COMPANY	)	ASLBP No. 00-771-01-LA
	)	
(Millstone Nuclear Power Station,	)	
Unit No. 3)	)	

NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON CONTENTIONS 4, 5 AND 6

INTRODUCTION

Pursuant to 10 C.F.R. § 2.1113, the Nuclear Regulatory Commission staff (Staff) hereby submits its written presentation summarizing all the facts, data and arguments of which the Staff is aware and on which the Staff intends to rely at oral argument, scheduled for July 19, 2000. For the reasons set forth below, the Staff submits that there is no genuine and substantial dispute of fact or law relating to Connecticut Coalition Against Millstone (CCAM) and Long Island Coalition Against Millstone (CAM) (collectively "Intervenors") Contentions 4, 5 and 6. This written presentation is supported by the affidavits of Anthony C. Attard, Laurence I. Kopp, James C. Linville and Antone C. Cerne.

BACKGROUND

On March 19, 1999, Northeast Nuclear Energy Company (NNECO or Licensee) filed an application for a license amendment, pursuant to 10 C.F.R. § 50.90, for the Millstone Nuclear Power Station, Unit No. 3 (Millstone 3). Letter to United States Nuclear Regulatory Commission from R.P. Necci, Vice President - Nuclear Oversight and Regulatory Affairs, Millstone Nuclear Power Station, Northeast Nuclear Energy Co., March 19, 1999.

(Application). *See* Exhibit 1. The Application sought approval to increase spent fuel storage capacity by installing two types of additional higher density spent fuel racks into the spent fuel pool. Exh. 1 at 1. On September 7, 1999, the NRC published a notice of proposed no significant hazards consideration determination and opportunity for hearing.[1] On October 6, 1999, Intervenors filed a request for hearing and petition to intervene.[2] The petition to intervene was granted by the Atomic Safety and Licensing Board (Board) by Memorandum and Order, dated February 9, 2000. *Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 3), LBP-00-2, 51 NRC 25 (2000).[3] The Board admitted three of the Intervenors' contentions for litigation. *Id.* at 32-41.

On February 22, 2000, pursuant to 10 C.F.R. § 2.1109, the Licensee filed a request to invoke the hybrid hearing procedures of Subpart K and for oral argument.[4] On April 19, 2000, the Board issued a memorandum and order acknowledging that this matter would proceed pursuant to Subpart K and establishing a schedule for filing written presentations

[1] "Northeast Nuclear Energy Company, et al.; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination and Opportunity for a Hearing," 64 Fed. Reg. 48,672 (1999).

[2] Petition to Intervene, October 6, 1999.

[3] In an earlier, unpublished opinion, the Board permitted the Intervenors to amend the petition, due to the Petition to Intervene's failure to adequately address standing. Memorandum and Order (Intervention Petition), October 28, 1999. A supplemental petition was filed on November 17, 1999. Supplemental Petition to Intervene in Behalf of Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone, November 17, 1999 (Supplemental Petition to Intervene). Responses were filed by NNECO and the Staff, on November 30, 1999 and December 7, 1999, respectively, and a prehearing conference was conducted on December 13-14, 1999.

[4] Northeast Nuclear Energy Company's Request for Subpart K Oral Argument, February 22, 2000.

and a date for oral argument.[5] In accordance with the Board's Order and 10 C.F.R. § 2.1113, the Staff hereby submits its written summary of all the facts, data, and arguments known to the Staff and on which the Staff intends to rely at oral argument to refute the existence of a genuine and substantial dispute of fact as to Contentions 4, 5 and 6.

STATEMENT OF FACTS

Millstone Nuclear Power Station, Unit 3 (Millstone 3) is a four-loop Westinghouse pressurized water reactor (PWR) operated by NNECO in New London County, Connecticut. The initial operating license was issued January 31, 1986, and expires November 25, 2025. The site on which Millstone 3 is situated contains two other nuclear power plants, Millstone Unit 1, a General Electric design boiling water reactor, which has been shut down since 1995 and is currently decommissioning, and Millstone Unit 2, a Combustion Engineering design PWR.

The Millstone 3 spent fuel pool (SFP) is located in the southwestern quadrant of the fuel building and is designed to accommodate fuel racks that store both spent fuel and fresh fuel assemblies. The Millstone 3 SFP was designed to hold up to 2169 fuel assemblies; however, at the time Millstone 3 was licensed, the licensed storage capacity was 756 fuel assemblies. Application, Exh. 1, Att. 3, at 3.

NNECO submitted a proposed license amendment request dated March 19, 1999 (as supplemented by letters dated April 17, 2000 , May 5, 2000 and June 16, 2000) to increase the licensed storage capacity from 756 to 1860 fuel assemblies, which will, according to the Application, provide capability to maintain a full core off-load and provide sufficient licensed capacity to allow operation near to the license expiration date in the year

[5] Memorandum and Order (Schedules for Proceeding), April 19, 2000.

2025. Exhibits 1, 17, 37, 38. Specifically, the proposed license amendment will revise the Technical Specification (TS) definition for spent fuel pool storage patterns (Section 1.40, Exh. 1, Att. 1, at 1), TS 3/4.9, "Refueling Operations; limiting condition for operations" and surveillance requirements for the subsections associated with boron concentration, spent fuel pool reactivity, the spent fuel pool storage pattern, and crane travel in spent fuel storage areas. Exh. 1, Att. 1, at 1-2. In addition, the proposed license amendment will revise TS 5.6, "Fuel Storage," to identify nominal center-to-center distances between fuel assemblies in the racks for the three regions of the SFP, fuel burn-up limitations, fuel enrichment limits, and SFP capacity by region. Exh. 1, Att. 1, at 3. The Licensee proposes to re-rack the Millstone 3 SFP by installing two types of higher density spent fuel racks into the SFP. Exh. 1, Att. 3, at 1. The proposed additional racks will have a closer assembly to assembly spacing to help maximize fuel storage capability. *Id.*

The planned SFP storage expansion involves the placement of 15 new rack modules into the Millstone 3 SFP. *Id.* The expansion will leave in place all 21 existing spent fuel racks that are in the Millstone 3 SFP. *Id.* After the expansion, the SFP will contain three distinct administratively controlled storage regions. *Id.* The NRC staff performs a safety review of the thermal hydraulic, structural, nuclear criticality and radiological aspects of the proposed actions described in the submitted amendment.

The proposed amendment was noticed in the Federal Register on September 7, 1999. *See* 64 Fed. Reg. 48,672 (1999).

THE REGULATORY FRAMEWORK

A. Subpart K, 10 C.F.R. § 2.1101, et seq.

This proceeding is governed by the hybrid hearing procedures of 10 C.F.R.

§ 2.1101 *et seq.* (Subpart K). Subpart K provides that its procedures may be used, at the request of any party, in contested proceedings concerning, *inter alia,* applications for a license amendment "to expand the spent fuel capacity at the site of a civilian nuclear power plant, through the use of high density fuel storage racks . . . ." 10 C.F.R. § 2.1103. [6] The procedures include a 90 day discovery period, followed by submittal of a detailed written presentation, and fifteen days thereafter, oral argument. 10 C.F.R. §§ 2.1111, 2.1113. The detailed written presentation must contain all the facts, data, and arguments known to the party and on which the party intends to rely at oral argument to support or refute the existence of a genuine and substantial dispute of fact. 10 C.F.R. § 2.1113(a). All supporting facts and data must be submitted in the form of sworn written testimony or other sworn written submission. *Id.* The written submissions are to be simultaneously served on all other parties. *Id.*

After considering the submissions and the oral arguments, the presiding officer will issue an order (1) designating any disputed issues of fact and law for hearing, and (2) disposing of any issues of fact or law not designated for hearing. 10 C.F.R. § 2.1115(a). In designating issues for hearing, the presiding officer "shall identify the specific facts that are in genuine and substantial dispute, the reason why the decision of the Commission is likely to depend on the resolution of that dispute, and the reason why an adjudicatory

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[6] Subpart K was promulgated in order to implement Section 134 of the Nuclear Waste Policy Act of 1982 (NWPA). Pub. L. 97-425, January 7, 1983, 96 Stat. 2201, 42 U.S.C. § 10101. *See* Hybrid Hearing Procedures for Expansion of Onsite Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors, Proposed Rule, 48 Fed.Reg. 54,490 (1983). *See also Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-00-12, 51 NRC ___, slip op. at 10-11 (May 5, 2000).

hearing is likely to resolve the dispute." *Id.* As for the issues not designated for hearing, only a brief statement of the reasons for the disposition is required. *Id.*

Subpart K provides for a form of summary disposition procedure. 50 Fed. Reg. 41,662, 41,664 (1984). There are several differences between the provisions of Subpart K and traditional NRC summary disposition practice, including: simultaneous filing of pleadings; mandatory oral argument; and placing the burden of demonstrating the *existence* of a genuine and substantial issue of material fact on the party requesting adjudication. *Id.* at 41,667. *Compare Georgia Power Co.* (Vogtle Generating Plant, Units 1 & 2), ALAB-872, 26 NRC 127 (1987).

In promulgating Subpart K, the Commission discussed the criteria for designating an issue for hearing, stating that:

> Not only must there be a genuine and substantial dispute of fact, but the dispute must be material: *i.e.,* the decision must be likely to depend on resolution of the dispute. In addition, the dispute must be one that can be resolved with sufficient accuracy only by the introduction of evidence in an adjudicatory proceeding.

50 Fed. Reg. at 41,666-67. The Commission also emphasized that the threshold for an adjudicatory hearing is strict:

> As the Commission pointed out in connection with the proposed rules, the statutory criteria are quite strict and are designed to ensure that the hearing is focused exclusively on real issues. They are similar to the standards under the Commission's existing rule for determining whether summary disposition is warranted. They go further, however, in requiring a finding that adjudication is necessary to resolution of the dispute and in placing the burden of demonstrating the existence of a genuine and substantial dispute of material fact on the party requesting adjudication.

Id. at 41,667. Therefore, in this case, the burden of going forward and of demonstrating the existence of a *genuine* and *substantial* issue of material fact[7] that can only be resolved by the introduction of evidence at an adjudicatory hearing[8] is on the Intervenors, CCAM and CAM. *See, e.g., Shearon Harris*, LBP-00-12, slip op. at 11. In order for any issue to proceed to hearing, the Board must "identify the specific facts that are in genuine and substantial dispute, the reason why the decision of the Commission is likely to depend on the resolution of that dispute,[9] and the reason why an adjudicatory hearing is likely to resolve the dispute."[10] 10 C.F.R. § 2.1115(a).

B. Prevention of Criticality in Spent Fuel Pools - General Design Criterion (GDC) 62

In their contentions, the Intervenors assert, *inter alia,* that criticality prevention for the spent fuel pool, as proposed in the Application, is inadequate. The asserted basis for Contention 6 is that GDC 62 prohibits the use of ongoing administrative measures, and

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[7] A pure issue of law will not require an adjudicatory hearing and should be decided on the briefs or oral argument. *See, e.g.,* 10 C.F.R. § 2.714(e).

[8] Even if the dispute is substantial and genuine, a hearing is unwarranted unless the dispute can only be resolved by the introduction of evidence at a hearing. In NRC practice, "evidence" would consist largely of documentary evidence, which has already been produced in this proceeding, and cross-examination of witnesses based upon written testimony, also already produced in connection with oral argument.

[9] This criterion is far stricter than a finding that an issue is material pursuant to the summary disposition rules. *See* 10 C.F.R. § 2.749(d).

[10] If an issue is designated for hearing, the governing procedures are found in Subpart G of 10 CFR Part 2, as modified by Subpart K. 50 Fed. Reg. at 41,664. "However, because discovery would precede the oral argument, there would ordinarily be no need for further discovery prior to the adjudicatory hearing. Accordingly, . . . the adjudicatory phase of the hearing would begin expeditiously after the presiding officer designated the issues meeting the criteria in 10 C.F.R. § 2.1106." *Id. See also* Shearon Harris, LBP-00-12, slip op. at 20 n.7.

NNECO's proposal to use credit for enrichment, burnup, and decay time are , according to the contention, ongoing administrative measures used to prevent criticality in the pool.

With respect to criticality, the NRC staff reviewed the Application to determine if it satisfied the requirements of 10 C.F.R. Part 50, Appendix A, Criterion 62 . GDC 62 provides:

> *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.

10 C.F.R. Part 50, Appendix A.

An implied basis for Contention 5 is that NNECO may not take credit for soluble boron in certain abnormal and accident conditions. Credit for soluble boron in certain conditions is permitted by the double contingency principle. The double contingency principle, as set forth in draft Regulatory Guide 1.13, revision 2, provides the analytical foundation for the Staff's analysis of criticality in spent fuel pools:

> 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 <u>not</u> occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 1.13, "Proposed Revision 2 to Regulatory Guide 1.13, 'Spent Fuel Storage Facility Design-Basis,'"Dec. 1981, at 1.13-14 (emphasis in original)(RG 1.13) (Exhibit 29). The principle, as implemented by the Staff, is articulated in 10 C.F.R. § 72.124, which states:

> 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.

The standard in Section 72.124, while not directly applicable to the matter before this Board, is similar to the guidance enunciated in RG 1.13, differing only in that it specifies that "sequential," as well as concurrent changes be considered in the evaluation.

THE ADMITTED CONTENTIONS

In its February 9, 2000 Order, the Licensing Board admitted three contentions for litigation in this proceeding. *Millstone 3*, LBP-00-2, 51 NRC at 32-41. All three contentions concern criticality.

Contention 4, "Undue and Unnecessary Risk to Worker and Public Health and Safety," states, as admitted:

> 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.

The Board stated that the contention has a two pronged basis: (1) "[T]he application contains a complex array of administrative controls;" and (2) "based on past experience, NNEC's ability to carry out such controls successfully is suspect." *Id.* at 32-33.

Contention 5, " Significant Increase in Probability of Criticality Accident," as admitted, reads:

> Will the proposed change in schedule of surveillance of the soluble boron in the fuel pool lead to a significantly increased likelihood of a criticality accident stemming from a misloaded fuel element, during the interval between fuel movements?

Id. at 36. The Board characterized this contention as a factual issue.

Contention 6, "Proposed Criticality Control Measures Would Violate NRC Regulations," reads, as admitted:

> Does GDC 62 permit a licensee to take credit in criticality calculations for enrichment, burnup, and decay time limits, limits that will ultimately be enforced by administrative controls?

Id. at 41. The Board characterized this contention as a legal issue, stating: "except with respect to identifying the precise administrative controls proposed to be utilized, as well as the existing administrative controls that would be superseded, the litigable issue posed by Contention 6 boils down to a question of law. . . ." *Id.*

<u>FACTS RELIED UPON BY THE STAFF</u>

The facts upon which the Staff relies with respect to Contentions 4, 5 and 6 are set forth in the affidavits of Drs. Laurence I. Kopp and Anthony C. Attard (Kopp/AttardAffidavit), James C. Linville (Linville Affidavit) and Antone C. Cerne (Cerne Affidavit), and the Exhibits submitted herewith. The professional qualifications of the Staff affiants are summarized in their affidavits and set forth in detail in attachments to each affidavit. Dr. Kopp and Dr. Attard are qualified as experts on criticality by virtue of their education, experience, and demonstrated knowledge and skill regarding that subject. Kopp/Attard Aff. ¶¶ 1,2. Mr. Linville was, until recently, Acting Director of the Millstone Project Directorate in Region I. Linville Aff. ¶ 1. Mr. Cerne is Senior Resident Inspector at Millstone 3. Cerne Aff. ¶ 1.

A summary of the facts presented in the affidavits is set forth below.

A. <u>Criticality and Reactivity</u>

Criticality is the achievement of a self-sustaining nuclear chain reaction. Kopp/Attard Aff. ¶ 6. The chain reaction proceeds as atoms of a fissile material absorb slow (thermal) neutrons and split (fission) into new light atoms (*i.e.*, fission products) and additional neutrons that, in turn, interact with additional fissile atoms. *Id.* Neutrons resulting from fission have high energy and are called "fast" neutrons. *Id.* Fast neutrons are not readily captured in U-235, the fissile material originally present in fresh fuel. Rather, a neutron must lose energy and "slow down," or become "thermalized" (a thermal neutron), in order to be readily captured in U-235 and cause fission. *Id.*

In order for fast neutrons to slow down, they must collide with, and transfer energy to, atoms. *Id.*, ¶ 7. This process is called "moderation." *Id.* A light element (such as hydrogen) is an effective moderator because the mass of its nucleus is on the same order as that of a neutron. *Id.* Therefore, upon initial collision, the neutron imparts most of its energy to the hydrogen nucleus and becomes thermalized. *Id.* Water, with its high hydrogen content, is the moderator in a light water reactor (LWR) such as Millstone. *Id.*

After being created through fission, during the process of moderation, and after reaching thermal energy levels, a neutron may undergo several events. *Id.*, ¶ 8. It may be absorbed by nonproductive capture in the fuel, the moderator, or the structural materials. *Id.* It may leak from the reactor system and either be reflected back into the system or be lost. *Id.* Finally, it may be absorbed by the U-235, cause fission, and produce more fast neutrons. *Id.*

When the process continues on its own, the system of atoms of fissile material is said to be critical. *Id.*, ¶ 9. The measure of criticality is the effective neutron multiplication factor, k-effective, or k_{eff}. *Id.* The multiplication factor is the ratio of the rate of neutron production to neutron loss due to fission, nonproductive capture, and leakage. *Id.* Criticality is achieved when k_{eff} is equal to 1.0. *Id.* When k_{eff} is less than 1.0, the system is subcritical. *Id.* Criticality can only occur in an array of LWR fuel if sufficient fissile material is available in a near-optimum geometry and a moderator (water) is present. *Id.* No array of LWR fuel can achieve criticality without water moderation present in the array. *Id.* Well-developed mathematical models (equations) exist in present-day computer codes and are used to compute k_{eff}. *Id.*

"Reactivity" is defined as $(k_{eff} - 1)/k_{eff}$. *Id.*, ¶ 10. When fuel is irradiated in a reactor as a result of operation and power generation, the reactivity of the fuel decreases over the design life of the fuel assembly. *Id.* This reduction of reactivity with irradiation is called "burnup." *Id.* Burnup is caused by the change in fissile content of the fuel (*i.e.*, depletion of U-235 and production of Pu-239 and other fissile actinides), the production of actinide absorbers, and the production of fission product neutron absorbers. *Id.* Before each reactor operating cycle, a licensee performs a reload analysis that predicts the burnup of each fuel assembly during the cycle. *Id.* These calculations are confirmed during the cycle by measurements of various operating characteristics, such as boron concentration and power distribution. *Id.* After every operating cycle (typically 1 to 2 years), approximately 1/3 of the fuel in a reactor is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment. *Id.* This irradiated (or spent) fuel is generally placed in a spent fuel pool at the reactor site and is replaced in the reactor by fresh (unirradiated) fuel. *Id.*

B. <u>Contention 4</u>

Licensees have used administrative procedures in essentially all burnup-dependent storage pools since the early 1980s. Kopp/Attard Aff. ¶ 13. These procedures generally consist of verification that the licensee has selected a fuel assembly that has zero burnup (new fuel), or assemblies that have achieved the required amount of burnup, based on plant operating records, and the licensee has stored it in the intended position in the spent fuel pool. *Id.* Administrative procedures are simply mechanisms for verifying physical processes and implementing physical controls. *Id.* Section 4.2.1 of American National Standards Institute (ANSI) standard ANSI/ANS-8.1-1983 states that nuclear criticality safety may be achieved by controlling one or more parameters of the system within subcritical limits and that control may be exercised administratively through procedures. *Id;* ANSI, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, Oct. 1983 (Exhibit 16). The NRC endorsed ANSI/ANS-8.1.1983 in revision 2 to Regulatory Guide 3.4. Kopp/Attard Aff. ¶ 13; U.S. Nuclear Regulatory Commission, Regulatory Guide 3.4, Revision 2, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuel and Materials Facilities," Mar. 1986 (Exhibit 48).

The Commission's regulations in 10 C.F.R. § 50.68 allow the use of administrative controls to prevent inadvertent criticality in fuel handling and storage. Kopp/Attard Aff. ¶ 14. The Commission developed 10 C.F.R. § 50.68 to allow holders of a construction permit or operating license for a nuclear power reactor issued under 10 C.F.R. Part 50 relief from the 10 C.F.R. § 70.24 requirement to maintain a criticality accident monitoring system in each area where nuclear fuel is handled, used, or stored, if criticality is precluded in these areas. *Id.* Specifically, 10 C.F.R. § 50.68(b)(1) allows a licensee to rely upon plant procedures to "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 conditions feasible by unborated water." *Id.* Sections (b)(2) and (b)(3) of 10 C.F.R.§ 50.68 allow licensees to use administrative controls or design features or both to prevent accidental flooding of new fuel racks to preclude criticality. *Id.*

NRC regulations allow the use of administrative controls to prevent criticality of fuel in storage. Kopp/Attard Aff. ¶ 15. Nothing in the applicable regulations makes a distinction between one time and ongoing administrative controls. *Id.,* ¶¶ 15, 18. Since human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. *Id.*, ¶ 15. To date, there have been no reported incidents of inadvertent criticality in U.S. spent fuel pools for any reason, including violation of administrative procedures. *Id.*

Millstone 3 currently incorporates administrative controls for two-region storage in the existing spent fuel storage racks. Kopp/Attard Aff. ¶ 16. These include fuel burnup/enrichment limitations as in Millstone 3 Technical Specification (TS) 3.9.13, Figure 3.9-1. *Id.;* Exh. 1, Att. 1, at 14. TS 3.9.13 and 3.9.14 require surveillance to ensure that all fuel assemblies are properly placed to maintain k_{eff} of the spent fuel pool less than or equal to 0.95 at all times. Kopp/Attard Aff. ¶ 16. These administrative controls are typical of those discussed above. The administrative controls proposed in the Application and in an April , 17, 2000 supplement to the Application only serve to augment the current procedures to the extent necessary to accommodate the 15 new storage racks and the changes in the regions. *Id.*, ¶ 16; *see* Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98), Apr. 17, 2000 (Exhibit 17). Millstone Unit 3 Surveillance Procedure SP 3866 and Millstone Nuclear Power Station Surveillance Procedure SP 31022 are detailed procedures designed to ensure and maintain appropriate boron concentration in the pool, and provide the steps required to ensure spent fuel pool k_{eff} remains less than or equal to 0.95 at all times. Kopp/Attard Aff. ¶ 16; Millstone Nuclear Power Station Surveillance Procedure SP 3866, Rev. 3, “Spent Fuel Pool Boron Concentration,” Nov. 8, 1996 (Exhibit 18); Millstone Nuclear Power Station Surveillance Procedure SP 31022, Rev. 4, “Spent Fuel Pool Criticality Requirements,” Apr. 20, 1997 (Exhibit 19). These administrative controls have not resulted in any reportable instances of fuel assembly misplacements at Millstone Unit 3. Kopp/Attard Aff. ¶ 16.

The above mentioned administrative procedural documentation suffices to ensure that adequate steps are taken to ensure that k_{eff} remains less than or equal to 0.95 (5% subcritical) for all regions of the Millstone 3 spent fuel pool. Kopp/Attard Aff. ¶ 18. Therefore, the new set of administrative controls does not pose any risk of a criticality accident.

Reliance on administrative controls is a given assumption in the safe operation of any nuclear power facility. Cerne Aff. ¶ 8; *See* 10 C.F.R. Part 50, Appendix B, Criterion V. These administrative controls are no greater or more complicated than those required for current operation of the Millstone Unit 3 spent fuel pool. Cerne Aff. ¶ 5. Section 6.8.1 of the Millstone Unit 3 Technical Specifications states that written procedures shall be established, implemented and maintained for refueling operations. *Id.*; Millstone Unit 3 TS 6.8.1., “Administrative Controls” (Exhibit 9). Section 13.5 of the Millstone Unit 3 FSAR requires plant procedures, including administrative procedures, to control the specifics of station operations. Cerne Aff. ¶ 5; Millstone Unit 3 FSAR Section 13.5 (Exhibit 10).

Licensee compliance with these procedures is a requirement of 10 C.F.R. Part 50, Appendix B, Criterion V. Cerne Aff. ¶ 5.

The December 1997 Notice of Violation and Proposed Imposition of Civil Penalties relied upon by the Intervenors does not include spent fuel pool violations at Millstone 3. Cerne Aff. ¶ 6; Notice of Violation and Proposed Imposition of Civil Penalties - $2,100,000 - NRC Inspection Report Nos. 50-245/50-336/50-423: 95-44, 95-82, 96-01, 96-03, 96-04, 96-05, 96-06, 96-08, 96-09, 96-201 (Dec. 10, 1997)(Exhibit 11). The two incidents documented in a 1994 Plant Information Report and in a 1995 Adverse Condition Report provided by NNECO in response to Intervenors' interrogatories and requests for document production concerning "errors" at the three Millstone facilities are the only two personnel errors documented with respect to Unit 3 spent fuel movement issues. Cerne Aff. ¶ 6; Plant Information Report No. 3-94-079, "Fuel Misplacement," (Jan. 14, 1991)(Exhibit 12); Adverse Condition Report Transmittal Sheet, ACR # 710, "SFP Crane Operator Went to Wrong Location [;] Stopped by Checker," (Apr. 27, 1995)(Exhibit 13). Both errors were identified during the spent fuel movement process and corrected before any assemblies were physically stored in an incorrect location. Cerne Aff. ¶ 6; Report No. 3-94-079, Exh. 12, at 3; ACR #710, Exh. 13, at 3-4. These incidents occurred prior to the 1996 Unit 3 shutdown and before the initiation of the recovery process that led to a significant Millstone Station culture change and improvements that permitted the Commission's authorization of the restart of Unit 3 in 1998. Cerne Aff. ¶ 6.

For both the existing and "new" spent fuel pool rack configurations, the proposed boron concentration in the spent fuel pool would effectively preclude the possibility of a criticality accident caused by a fuel assembly misplacement. Cerne Aff. ¶ 4; *See* Millstone

Nuclear Power Station, Unit No. 3 Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98) (Apr. 17, 2000)(Exhibit 17).

In response to inaccuracies in the Millstone 1 Updated Safety Analysis Report, a Licensee root cause analysis documented a programmatic breakdown in the configuration management controls at Millstone 1 and acknowledged that the potential existed for configuration management problems at the other two units. Linville Aff. ¶ 6. Further Licensee findings and NRC inspections identified design control deficiencies and degraded and nonconforming conditions at Millstone 2 and 3, which included (1) errors in the licensing and design bases documentation; (2) failures to translate design bases into procedures and hardware; and (3) inadequate engineering and modifications. *Id.* These conditions led the NRC to issue letters requiring that, before restarting each unit, the licensee inform the NRC of the corrective actions taken regarding design configuration issues at Millstone Units 2 and 3. *Id.*

In June 1996, the NRC designated the three units as Category 3 plants on the NRC's Watch List. Linville Aff. ¶ 7. Plants in this category, which required Commission authorization to resume operation, had significant weaknesses that warranted maintaining them in a shutdown condition until the licensee could demonstrate to the NRC that it had both established and implemented adequate programs to ensure substantial improvement. *Id.* Several NRC initiatives were directed toward assuring correction of existing problems prior to restart of the units. *Id.*, at ¶¶ 8-10. In SECY-97-003, the Staff described to the Commission processes and approaches that the Staff would use to oversee the corrective action programs at Millstone 1, 2, and 3. Linville Aff. ¶ 10; SECY-97-003, "Millstone Restart Review Process," Jan. 13, 199, at 3-11 (Exhibit 3). The Staff applied the guidelines of NRC

Inspection Manual Chapter 0350, "Staff Guidelines for Restart Approval" (Exhibit 4) to the restart approvals for Millstone 1, 2, and 3, which included developing a Restart Assessment Plan for each unit to consolidate all of the NRC's restart issues. Linville Aff. ¶ 10; SECY 97-003, Exh. 3, at 4.

Procedure quality and adherence had been a chronic problem at Millstone since the early 1990s. Linville Aff. ¶ 11; Restart Assessment Plan, Exh. 3, Att. 1, at 8. The licensee's Procedure Enhancement Program was identified as a "significant item" in the Restart Assessment Plan. Linville Aff. ¶ 11; Restart Assessment Plan, Exh. 3, Att. 1, at 5. NRC undertook a series of inspections, including inspections of the Procedure Upgrade Program, the NRC Independent Corrective Action Verification Program, and an Operational Safety Team Inspection in the 1996-1998 time frame. Linville Aff. ¶ 12. These inspections found that there had been a substantial improvement during those two years and that Unit 3 procedures were acceptable for restart. Linville Aff. ¶ 13. In June 1998, the Commission authorized restart, the Watch List status of Unit 3 was changed from Category 3 to Category 2, and the Executive Director of Operations was named the senior manager responsible for approving commencement of actions to restart Unit 3. Linville Aff. ¶ 13. The Executive Director for Operations authorized the utility to commence actions to restart Unit 3, and the reactor started up on June 30, 1998. *Id.*; Letter, L.J. Callan, NRC, to B. Kenyon, NNECO (June 29, 1998)(Exhibit 41).

At the NRC's April 1999 Senior Management Meeting, the NRC determined that Unit 3 had not yet demonstrated sustained, successful plant performance and that Unit 3 warranted oversight as a regional-focus plant. Linville Aff. ¶ 14; Letter, J.C. Linville, NRC, to R.P. Necci, NNECO, “Mid-Cycle Plant Performance Review - Millstone Station” (Sept. 30, 1999)(Exhibit 36).

On March 9, 1999, the Commission approved the Staff’s recommendation to close an Order of October 24,1996, that required third-party oversight of Millstone employees’ safety concerns, given the licensee’s improved performance in that area. Linville Aff. ¶ 17. *See* SECY-99-10, “Closure of Order Requiring Independent, Third-Party Oversight of Northeast Nuclear Energy Company’s Implementation of Resolution of the Millstone Station Employees’ Safety Concerns” (Jan. 12, 1999)(Exhibit 46); Staff Requirements Memorandum - SECY-99-010 (May 25, 1999)(Exhibit 47).

Millstone Unit 3 was removed from regional focus and returned to normal NRC oversight following the May 2000 Senior Management Meeting. Linville Aff. ¶ 18. The basis for the decision was presented to the Commissioners at a meeting on May 25, 2000. *Id.* Although criteria for returning to normal NRC oversight of employee safety concerns have been met, the NRC will continue to follow licensee and third party activities in this area within the normal processes of the new Reactor Oversight Program. Linville Aff. ¶ 18.

C. Contention 5

Included in NNECO’s request for license amendment was a TS change that would modify TS 3.9.1.2 to require surveillance of the spent fuel pool boron concentration only during times of fuel movement within the spent fuel pool. Kopp/Attard Aff. ¶ 21; Exh. 1, Att. 1, at 1. The minimum required boron concentration would have been changed from its present value of greater than or equal to 1750 parts per million (ppm) (required to account for possible Boraflex degradation), to 800 ppm, which is more than sufficient to maintain k_{eff} less than or equal to 0.95 for the worst misloading event. Kopp/Attard Aff. ¶ 21; Exh.

1, Att. 3, at 10.[11] TS 4.9.1.2 would have been changed to require monitoring of the boron concentration every seven days only during fuel movement, rather than every 72 hours when fuel assemblies were in the pool. Kopp/Attard Aff. ¶ 21; Exh. 1, Att. 1, at 1.

One of the issues raised by CCAM/CAM was that NNECO's proposal to monitor soluble boron only during fuel movements eliminated a barrier against criticality at Millstone Unit 3. Kopp/Attard Aff. ¶ 22.

In an April 17, 2000 letter to the NRC, NNECO stated that after evaluation of the outcome of the February 9, 2000 prehearing conference, it had decided to change the proposed TS amendment to require, in TS 3.9.1.2, that a proposed boron concentration of 800 ppm be maintained at all times when fuel assemblies are stored in the spent fuel pool. Kopp/Attard Aff. ¶ 23; Exh. 17 at 1-2. TS 4.9.1.2 would be amended to require verification of the boron concentration every 7 days. Kopp/Attard Aff. ¶ 23; Exh. 17 at 2.

NNECO's criticality analysis, prepared by Holtec International for Millstone Unit 3, included several conservative assumptions. Kopp/Attard Aff. ¶ 24; Holtec International, Licensing Report for Spent Fuel Rack Installation of Millstone Nuclear Station Unit 3 (Non-proprietary version), Exh. 1, Att. 5, at 4-2. The analysis assumed the presence of unborated water. *Id.* The analysis contained several other conservative assumptions. *Id.* Racks were assumed to contain the most reactive fuel authorized to be stored in the facility. *Id.* Unborated water was assumed to be at the temperature yielding the highest reactivity over the expected range of water temperatures. *Id.* The analysis assumed an infinite array

[11]Since the Boraflex in the existing storage racks is not being credited in the Licensee's criticality safety analysis, the requirement for 1750 ppm of boron to account for possible Boraflex degradation is no longer necessary. Kopp/Attard Aff. ¶ 21. Application, Exh. 1, Att. 3, at 10.

of storage cells, that is, no neutron leakage (except when assessing peripheral effects and certain accidents). *Id.* The analysis neglected neutron absorption by minor structural material. *Id.* In order to maximize the calculated k_{eff}, the analysis incorporated calculational uncertainties and biases, as well uncertainties due to manufacturing tolerances.[12] *Id.*

As part of the NRC review of the NNECO amendment request to establish three regions (Regions 1, 2,and 3) for fuel storage in the spent fuel pool, the staff reviewed the Holtec International report, which presented the criticality evaluation for the misloading of a fresh fuel assembly in the Millstone 3 spent fuel pool. Kopp/Attard Aff. ¶ 25. Based on the analysis performed by Holtec and described in this report, NNECO has determined that a soluble boron concentration of only 425 ppm would be sufficient to maintain a 5% subcriticality margin in the event of a fuel assembly misloading event (i.e., a fresh PWR assembly enriched to 5 weight-percent U-235 inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity). *Id.* NNECO conservatively chose a value of 800 ppm in the proposed TS. *Id.*; Exh. 1, Att. 1, at 1; Exh. 17 at 2. In addition, NNECO submitted additional Holtec analyses and calculations which demonstrated that criticality is precluded even for various beyond design basis combinations of fresh fuel misplacements and boron dilution. Kopp/Attard Aff.¶¶ 26, 29.

[12]In LBP-00-02, the Board stated, "According to CCAM/CAM, the evaluation submitted by NNEC clearly stated that a single movement error can result in the required criticality margin being violated unless there is soluble boron in the spent fuel pool water." 51 NRC at 35; *see* Supplemental Petition at 16-17. The Staff notes that, pursuant to the double contingency principle, the 5 % criticality margin need only be met for a single accident scenario. In addition, in the scenario here, which involves two unlikely, independent and concurrent failures the system remains subcritical. Thus, it is in conformance with GDC 62.

TS 3.9.1.1 for Millstone Unit 3 requires a minimum boron concentration of 2600 ppm in all filled portions of the reactor coolant system (RCS) and the refueling canal during refueling operations. Kopp/Attard Aff. ¶ 27; Millstone Unit 3 Technical Specification 3.9.1.1 (Exhibit 51). During refueling, the water volume in the spent fuel pool and the refueling canal form a single mass. Kopp/Attard Aff. ¶ 27. As a result, the soluble boron concentration is relatively the same in each of these volumes. *Id.* Therefore, the actual fuel pool boron concentration is approximately 2600 ppm during refueling operations. *Id.* As a practical matter, boron in the spent fuel pool does not disappear after fuel movements, nor is it appreciably diluted over time. *Id.* Any hypothetical event that could dilute the normal boron concentration in the Unit 3 spent fuel pool of approximately 2600 ppm by any significant amount would require such large quantities of water that it would be detectable well before the 800 ppm limit imposed by TS 3.9.1.2 was reached. *Id.* In addition, high and low water level alarms are present in the fuel handling area and in the control room which would alert the operators to an increasing or decreasing water level in the pool. *Id.*

RG 1.13 recommends that the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures. Kopp/Attard Aff. ¶ 28; RG 1.13, Exh. 29 at 1.13-14. This additional safety assurance is based on application of the "double contingency principle" as defined in ANSI/ANS-8.1-1983, which was endorsed by the NRC staff in a generic communication sent to all power reactor licensees on April 14, 1978. Kopp/Attard Aff. ¶ 28; ANSI/ANS-8.1-1983 (Exh. 16); Letter, B.K. Grimes, NRC, to All Power Reactor Licensees, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," Apr. 14, 1978 (Exhibit 27). More recently, the Commission affirmed this endorsement in 10 C.F.R.§

72.124(a), which requires at least two unlikely, independent, concurrent or sequential events to have occurred before a nuclear criticality accident is possible. Kopp/Attard Aff. ¶ 28. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered one unlikely accident condition and a second concurrent independent accident need not be assumed. *Id.* Alternatively, credit for the presence of soluble boron in PWR pools may be assumed in evaluating other accident conditions such as the misloading of fresh fuel or fuel that has not attained the required minimum burnup into the proposed region 1, 2, and 3 storage racks. *Id.*

On June 21, 2000, NNECO supplemented its responses to Intervenors' third set of interrogatories by submitting several calculations performed by Holtec International, of the K_{eff} for each of the three Regions in the pool, assuming various misplacements of fresh fuel assemblies in each Region. Kopp/Attard Aff.¶ 29; NNECO's Supplementary Response to CCAM and CAM's Third Set of Interrogatories (June 21, 2000). This analysis includes beyond-design basis criticality calculations involving fresh fuel assembly misplacements and boron dilution. *Id.* None of these scenarios resulted in criticality in the spent fuel pool. *Id.* Staff review of this document indicates that the calculated criticality values for each Region are correct.

D. Contention 6

NRC regulations require that licensees prevent criticality in their spent fuel pools. Kopp/Attard Aff. ¶ 33; 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 62. GDC 62 states 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." *Id.* A proposed version of the GDC was sent to the Commission in a paper dated June 16, 1967. Kopp/Attard Aff. ¶ 33; Memorandum from W.B. McCool to Atomic Energy Commission, "Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits," AEC-R 2/57, June 16, 1967 (Exhibit 22). The AEC (predecessor to the NRC) first formally published the general design criteria for comment on July 11, 1967. Iopp/Attard Aff. ¶ 33; Proposed rule, General Design Criteria for Nuclear Power Plant Construction Permits, 21 Fed. Reg. 10,213 (1967). At that time, the proposed criterion for prevention of fuel storage criticality was labeled GDC 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." Kopp/Attard Aff. ¶ 33.

The AEC received two public comments regarding Criterion 66. Kopp/Attard Aff. ¶ 34. The first comment was received from the Oak Ridge National Laboratory (ORNL) on September 6, 1967. *Id;* Letter, W.B. Cottrell, ORNL, to H.L. Price, Atomic Energy Commission, "Review of USAEC 'General Design Criteria for Nuclear Power Plant Construction Permits,' July 11, 1967," Sept. 6, 1967 (Exhibit 24). Specifically, the ORNL comment on proposed GDC 66 stated that ORNL did not understand the implication of "or processes" at the end of the first sentence, nor did they believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. *Id.* They suggested that the last sentence of the criterion should read as follows: "Such means as geometrically safe configurations shall be used to insure that criticality cannot occur." *Id.* The AEC staff considered these comments and decided that it was not necessary to change the phrase "or processes" and, therefore, it was retained. Kopp/Attard Aff. ¶ 34. In considering the second comment, the AEC staff felt that, although the assurance of geometrically safe configurations was the preferable means for preventing criticality, procedural controls should not be ruled out. *Id.* Therefore, GDC 66 (renumbered GDC 62) was revised to state that geometrically safe configurations are the preferable means for preventing criticality in fuel handling and storage. *Id.;* Memorandum, H.L. Price, AEC, to Commissioners, "Status Report on General Design Criteria," July 6, 1970 (Exhibit 25); Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)(Exhibit 26). However, it did not specifically rule out other means. Kopp/Attard Aff. ¶ 34.

The second comment was received from Atomics International. *Id.,* ¶ 35; Letter, J. Flaherty, Atomics International, to Secretary, Atomic Energy Commission, Sept. 25, 1967 (Exhibit 30). This comment suggested that the second sentence of the proposed criterion be replaced with "Inherent means should be used where practicable." *Id.* Although these specific suggested word changes were not incorporated in the final criterion, the AEC did incorporate the intent of the Atomics International comment by stating that geometrically safe configurations (inherent means) were preferred. Kopp/Attard Aff. ¶ 35.

Fuel reactivity is determined by its physical design, its initial (fresh) enrichment (or weight percent of U-235 to total uranium), and fuel depletion or burnup. Kopp/Attard Aff. ¶ 36. The initial enrichment of a fuel assembly is a physical process consisting of manufacturing an assembly containing a specified weight percent of U-235. *Id.*

Burnup credit is the practice of accounting for the reduced reactivity of spent fuel due to fissile material decay and fission product buildup described above in evaluating criticality safety. Kopp/Attard Aff. ¶ 37. The regulations do not elaborate on how or how much subcriticality should be assured, nor do they prohibit the use of burnup credit for criticality safety. *Id.* Burnup of fuel occurs as a natural consequence of the fuel's use in a reactor. *Id.* Therefore, fuel burnup is a physical process and credit for burnup to prevent criticality in spent fuel storage pools is permitted under GDC 62. *Id.* Decay time is an extension of the burnup process and includes the time an assembly has been discharged from the reactor and resides in the storage pool. *Id.* Spent fuel decay time results from the radioactive decay of fissile isotopes in the spent fuel (e.g., U-235) to non-fissile (or non-productive) neutron absorbing isotopes. *Id.* Thus, the additional decay while cooling in the spent fuel pool further reduces the fuel reactivity. This loss in reactivity due to decay time allows a reduction in the minimum burnup needed to meet the reactivity requirements and typically is applicable to older fuel that has been stored in the spent fuel storage racks for a period of years. *Id.*

The NRC has established a 5% subcriticality margin for wet storage of spent fuel assemblies to assure that licensees meet the requirements of GDC 62. Kopp/Attard Aff. ¶ 38. The NRC staff stated this acceptance criterion for criticality in a generic communication sent to all power reactor licensees. *Id;* Grimes letter, Exh. 27. This letter states that "The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions." Grimes letter, Exh. 27, at III-3. Section 5.6.1.1 of the Millstone Unit 3 technical specifications (TS) also contains this requirement, which states, "The spent fuel storage racks are made up of three regions which are designed and shall be maintained to ensure a k_{eff} less than or equal to 0.95 when flooded with unborated water." Kopp/Attard Aff. ¶ 38; Exh. 1, Att. 1, at 25.

NNECO proposes to use administrative procedures at Millstone Unit 3 to verify that a fuel assembly has achieved the required amount of burnup to be placed in the proposed storage racks of regions 1, 2, and 3. Kopp/Attard Aff. ¶ 39. To date, more than 50 plants have obtained NRC approval for the use of burnup credit for spent fuel storage. *Id.* The NRC first approved burnup credit in spent fuel pool storage analyses in the early 1980s. *Id.* Licensees have established their ability to predict core burnup behavior over hundreds of reactor years of operation. *Id.* They have also established the ability to predict isotopic inventories of reprocessed fuel from data available from several cores of the Yankee reactor. *Id; see* Abstract, R.J. Nodvik, "Evaluation of Mass Spectrometric and Radiochemical Analysis of Yankee Core I Spent Fuel," Mar. 1996 (Exhibit 28). Therefore, the NRC has allowed licensees to take credit for burnup in criticality analyses of spent fuel storage pools. Kopp/Attard Aff. ¶ 39.

Intervenors have stated that there are two classes of administrative measures: those that are made over a finite time and after having been made are no longer necessary (the so called "one time" measures); and administrative measures that are required on an ongoing basis. Kopp/Attard Aff. ¶ 18. There is nothing in the applicable regulations that supports this interpretation. *Id.*

The staff considers fuel misplacement in the Millstone 3 regions 1, 2, and 3 storage racks to be an unlikely event for several reasons. Kopp/Attard Aff. ¶ 40. First, proposed TS 3.9.13 will control fuel storage limitations and second, selection procedure SP 31022, described above, will control fuel assembly selection. *Id.;* SP 31022, Exh. 19. Therefore, both TS as well as plant procedures would have to be violated for a fuel assembly misplacement to occur. Kopp/Attard Aff. ¶ 40. In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. *Id.* Finally, the burnup limit curves (Figures 3.9-1, 3.9-3, and 3.9-4) proposed for the Millstone Unit 3 TS for safe storage in regions 1, 2, and 3, are based on a minimum required burnup. *Id;* Exh. 1, Att. 1, at 14, 16, 17. These are bounding values that result in just meeting the 5% subcriticality margin for storage pools in region 1, 2, and 3. Kopp/Attard Aff. ¶ 40. In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). *Id.* Such fuel assemblies, therefore, should fall in the acceptable burnup domain specified in these figures, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. *Id.* Consequently, the placement of a fuel assembly that does not meet the technical specification burnup requirements into a large storage pool, such as the Millstone 3 SFP, and the continued failure to detect this misplacement, is a highly unlikely event at Millstone 3. *Id.* Multiple misplacements are even more unlikely. It is highly unlikely that a single failure in the administrative controls or the management process will lead to misplacement or multiple misplacements. Such multiple misplacements, with or without boron dilution, leading to criticality, are highly improbable and well beyond the application of the double contingency principle. *Id.* Although there have been several reported fuel assembly misplacements in spent fuel pools at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in detecting and correcting fuel misloadings. *Id.* Further, there have been no reported incidents of boron dilution events occurring concurrently with fuel misloading events. *Id.*

ARGUMENT

A. Intervenors' Designated Expert Witness, Gordon Thompson , Should Be Disqualified As an Expert Witness and his Testimony/Declaration Stricken. The Testimony/Declaration of Intervenors' Second Designated Expert Witness, David Lochbaum, Should be Given No Weight.

The Intervenors have proffered Gordon Thompson as their expert witness for Contentions 4, 5 and 6, inadequate criticality control. The staff submits that the Intervenors have not demonstrated Dr. Thompson's expertise in criticality control or any other issue related to these Contentions.

It is well established that persons who seek to present expert testimony must be qualified to do so. In order to establish an expert witness's testimony as reliable, a party seeking to offer such a witness's expert opinion must show that the witness has the necessary qualifications to offer an expert opinion on the matter, or must be capable of questioning the soundness of the expert opinions of persons who do appear as witnesses in the proceeding. 10 C.F.R. § 2.733 (1999); *see, e.g., Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 & 2), ALAB-669, 15 NRC 453, 475 (1972). This standard is at the heart of the Commission's requirement of reliability.

While the Federal Rules of Evidence are not directly applicable to Commission proceedings, NRC adjudicatory boards often look to those rules for guidance, and have adopted the standard for expert witnesses enunciated in Rule 702 as allowing a witness to be qualified as an expert by "knowledge, skill, experience, training, or education" to testify "[i]f scientific, technical, or other specialized knowledge will assist the trier of fact to understand the evidence or to determine a fact in issue." *McGuire*, ALAB-669, 15 NRC at 475; *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2

& 3), ALAB-717, 17 NRC 346, 365 n.32 (1983); *Philadelphia Elec. Co.* (Peach Bottom Atomic Power Station Units 2 & 3), ALAB-701, 10 NRC 1517, 1524 (1982).

In the *McGuire* case, proposed expert testimony on hydrogen combustion and control was excluded as non-expert where the witness, a physical organic chemist with a master's degree, claimed to have an "ability to understand and evaluate matters of a technical nature," based on his "academic and practical training" and "years of reading AEC and NRC documents," but lacked specific expertise in the subject in issue. *McGuire*, ALAB-669, 15 NRC 453. *See also Pacific Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), LBP-87-25, 26 NRC 168, 182 (1987) (testimony of Ph.D. in physics with no experience with the technical subjects under review: nuclear engineering, nuclear systems, nuclear criticality, seismic design, etc., admitted but only given the weight appropriate considering his Ph.D. and years of teaching physics); *Philadelphia Elec. Co.* (Peach Bottom Atomic Power Station Units 2 & 3), ALAB-701, 16 NRC 1517, 1523-24 (1982) (Ph.D. in chemistry not qualified to give expert opinion on health effects of radon releases, due to lack of education or experience in medicine, health physics or other areas related to health effects of radon). *Cf. Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 and 4), LBP-86-23, 24 NRC 108, (1986) (testimony of Ph.D. in mathematics with no training or knowledge of engineering, heat transfer and other technical issues under review, permitted to testify and act as expert interrogator to "sharpen the issues").

The Intervenors have provided no evidence to demonstrate that Dr. Thompson is qualified to serve as an expert witness in this case. As demonstrated below, Dr. Thompson does not possess the "'knowledge, skill, experience, training, or education' germane to" the criticality issues under consideration in this proceeding. *McGuire*, ALAB-669, 15 NRC at 475.

In his deposition in connection with this case, held on May 10, 2000, Dr. Thompson stated that he would be providing, for the Intervenors, “an interpretation of what the intent was of the parameters of that criteria and the significance of criticality issues that would be pertinent to determining their intent . . . . “ Deposition of Gordon Thompson, Ph.D, *Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit No. 3), Docket No. 50-423-LA-3, May 10, 2000, at 4 (Thompson Deposition)(Exhibit 14). He admitted that he was not involved in the framing of GDC 62 and he has never commented on GDC 62. *Id.* He has done a preliminary qualitative analysis of the consequences of a criticality incident in relation to the Harris case. *Id.* at 12, 13. He has never been involved in moving fuel at a nuclear power plant, reviewing procedures related to fuel movements, or calculating criticality to determine K effective. *Id.* at 13-14.

Although not directly addressed in the deposition relating to the instant case, Dr. Thompson’s qualifications were recently addressed in the matter of Carolina Power & Light Co., (Shearon Harris Nuclear Power Plant), Docket No. 50-400-LA.[13] During his deposition in the Harris case, Dr. Thompson testified as follows:

> 1) He has a Ph.D. in applied mathematics;
>
> 2) He has no training in fission reactor engineering, fission reactor criticality control or fission reactor criticality analysis;

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[13] *See* NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon Which the Staff Proposes to Rely at Oral Argument on Technical Contentions 2 and 3, (January 4, 2000) at 14-19.

3) He claimed to be an expert in fission reactor criticality analysis for the purpose of the Harris proceeding based on his "basic expertise in scientific principles and analytic principles" and his "general experience with engineering in general and nuclear plant engineering in specifics." His contribution to that proceeding would rely upon the application of general scientific principles to the criticality contention);

4) He has no training in running criticality analysis codes and would not be running any codes in connection with that proceeding. He would confine his analysis to the assessment of the sufficiency of the assumptions which go into the analysis as to whether they address the issues of concern in connection with criticality. He will not address the analysis itself, because he is not competent to do so;

5) He has never been licensed as a nuclear power plant operator. He has no training or experience in the operation of a nuclear power plant, engineering at a nuclear power plant, or writing or implementing procedures at a nuclear power plant;

6) He is not an expert in nuclear power plant operations, but he claims to have "performed studies and presented testimony relating to the safety of nuclear facilities, including nuclear power plants; and in the course of those studies and preparing those testimonies, [he] has become expert in operational matters pertinent to the analyses and testimony. So in that limited sense, [he] is an expert in operations. It's a very circumscribed sense.";

7) He claims to be familiar with details of several nuclear facilities in several countries. He claims to have "always taken pains to acquire the necessary familiarity with the details of the design and operation of each facility in order to support whatever claim [he] made;"

8) He has also been in several fuel handling buildings for approximately an hour or so each;

9) None of the other projects he worked on, his publications, or his expert presentations and testimony have dealt with his analysis of assumptions used in criticality analysis.

Deposition of Gordon Thompson, Ph.D., *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), Docket No. 50-400-LA, Oct. 21, 1999, at 21-39 (Harris Deposition) (Exhibit 15).

Based on the sworn testimony of Dr. Thompson during his deposition in both the Harris and the instant case, as set forth above, it is clear that he lacks expertise related to any issue relevant to the admitted contentions. Moreover, by his own admission, none of the numerous “Project sponsors and tasks,” “Publications,” or “Expert testimony” listed in Dr. Thompson’s Curriculum Vitae relate to spent fuel pool criticality issues.[14] Harris Dep., Exh. 15 at 38-39. His only known prior input on criticality issues was in the Harris case.[15] Therefore, his Curriculum Vitae does not demonstrate that Dr. Thompson has sufficient knowledge, skill, experience, training or education related to the subject matter of the admitted contentions.[16]

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[14] Other than the recent Shearon Harris matter.

[15] The Staff made a similar motion to disqualify Dr. Thompson in the *Shearon Harris* case. *See* footnote 8. The Licensing Board in that case denied the motion stating: “After hearing party presentations regarding [the Staff’s motion to disqualify] during . . . oral argument . . . the Board ruled from the bench that it would not declare Dr. Thompson ineligible to be the BCOC expert on this matter, but would assign his testimony appropriate weight commensurate with his expertise and qualifications, . . . In this regard, we note that by reason of his experience and training, his expertise relative to reactor technical issues seems largely policy-oriented rather that operational.” *Shearon Harris*, LBP-00-12, slip op. at 30-31 n.9. Although the Staff does not agree that Dr. Thompson has any expertise relating to the technical issues under consideration herein, the Staff submits that the decision in Harris is persuasive authority that Dr. Thompson is, at most, qualified to render an expert opinion as to policy issues alone, and does not have the requisite expertise to render an opinion regarding operational issues.

[16] The Curriculum Vitae gives no indication of the issues Dr. Thompson addressed in these projects or his area of expertise applied to them. Nor was the outcome of any project or expert testimony indicated. Nor does it appear that any of his publications or (continued...)

In addition, should he be offered in support of the Intervenors' position regarding Contention 4 (the complexity of the proposed administrative measures), the Staff notes that there is nothing in Dr. Thompson's background to suggest any expertise in administrative procedures and the ability of workers to perform them. He has absolutely no training or experience in the area of human factors. Therefore, any written testimony or declaration he may provide in this area should be given no weight by the Board.

Similarly, Mr. Lochbaum, the Intervenors' other expert witness, has no training or experience in the area of human factors. Therefore, any written testimony or declaration he may offer in reference to Contention 4 regarding the increased "complexity" of the administrative should be given no weight by this Board.

The rationale for permitting expert testimony and permitting an expert to give conclusions and opinions based upon data and facts, and other experts' analyses, is to aid the trier of fact in reaching a decision. *See McGuire,* ALAB-669, 15 NRC at 475. In this proceeding, the Board includes members with technical backgrounds, training and experience exceeding Dr. Thompson's. Therefore, any opinions he may render in this matter based upon his "basic expertise in scientific principles and analytic principles," his "general experience with engineering in general and nuclear plant engineering in specifics," and his application of general scientific principles to the criticality contention will not aid the Board in rendering a decision on the criticality contentions. Harris Dep., Exh. 15, at 21-22; *see McGuire*, ALAB-669, 15 NRC at 475 n. 48.

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[16](...continued)
studies were subject to peer review. None of his listed publications, other than that related to the Harris case and this proceeding, concern any of the issues, data or facts involved herein.

Since the Intervenors have the burden of demonstrating that a genuine and substantial issue of material fact exists and that an evidentiary hearing is required to resolve the issues, as discussed above, it is imperative that they provide competent evidence to support their positions. Since Dr. Thompson lacks the expertise to provide competent evidence as to the issues herein, his testimony should be given no weight.

As demonstrated above, neither Dr. Thompson nor Mr. Lochbaum qualify as expert witnesses by virtue of their knowledge, skill, experience, training, or education. Therefore, any conclusions they make, opinions they render, or other testimony related to these contentions must be stricken.[17]

B. Contention 4

As noted above, Contention 4, as admitted by the Licensing Board, reads as follows:

> 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.

LBP-00-2, 51 NRC at 34. The Board adopted the "two-pronged" basis offered by Intervenors: (1) the application contains a complex array of administrative controls and (2) NNECO's ability to carry out such controls successfully is suspect, as reflected in a March 1996 issue of TIME Magazine and a December 1997 civil penalty/notice of violation, in which NNECO was cited for failure to maintain the plant's spent fuel pool configuration in

[17] Since the intervenor is not offering Dr. Thompson or Mr. Lochbaum as fact witnesses, the Board cannot merely limit their testimony to facts based on personal observation.

conformance with design and accident analyses performed by Holtec International. *Id.* at 32-33.

The first prong of Contention 4 is addressed by Drs. Kopp and Attard in their affidavit and by Mr. Cerne, Senior Resident Inspector at Millstone, in his. With regard to the alleged trade-off of reliance on physical protection for administrative controls, Drs. Kopp and Attard and Mr. Cerne demonstrate that this claim is without basis. The "new" administrative controls are no greater or more complicated than those required for current operation of the Millstone Unit 3 spent fuel pool. Cerne Aff. ¶ 4. Drs. Kopp and Attard point out that Millstone 3 currently incorporates administrative controls for two-region storage racks and these include fuel burn-up/enrichment limitations. Kopp/Attard Aff. ¶16. Millstone's administrative controls are typical of those used at other plants. *Id.* The "new" administrative controls only serve to augment the current procedures to the extent necessary to accommodate the 15 new racks and the changes in the regions. *Id.*

Intervenors have stated that there are two classes of administrative measures: those that are made over a finite time, and after completion are no longer necessary (the so-called "one time" measures); and administrative measures that are required on an ongoing basis. In reply to the NRC staff's first set of interrogatories, the Intervenors state that "(P)hysical protection against criticality may rely on one-time administrative measures to ensure that the physical protection is in place." Connecticut Coalition Against Millstone and Long Island Coalition Against Millstones Reply to NRC Staff's First Set of Interrogatories, at 9 (April 8, 2000). They further state, "(F)or example, if physical protection is provided by the geometry of racks, one-time administrative measures will be needed to ensure that the racks are constructed so as to preserve the specified geometry under specified conditions."

Id. As stated above, there is nothing in the applicable regulations that supports this theory. Intervenors provide no support or basis for this assertion and the Staff submits that it is without merit. The Staff does note, without accepting the Intervenors' theory concerning administrative measures to prevent criticality, that the selection of a fuel assembly for storage in region 1, 2, or 3 of the Millstone 3 spent fuel pool is a one-time administrative measure which, under Intervenors' definition, may be relied upon for physical protection against criticality. Thus, the "new" administrative controls do not trade off physical protection for administrative controls.

The first prong of Contention 4 also includes an assertion that the alleged trade-off of physical protection for administrative controls poses an undue risk of a criticality accident. This is erroneous. The proposed boron concentration for the spent fuel pool effectively precludes criticality in the event of an error that results in a misplaced fuel assembly. Cerne Aff. ¶ 4. Drs. Kopp and Attard address this matter in responding to Contention 5. They state that they have reviewed NNECO's submission to the Board and parties, dated June 16, 2000, which forwarded the calculations performed by Holtec International for the K-effective for each of the three regions in the pool assuming various misplacements of fresh fuel assemblies in each region. Kopp/Attard Aff. ¶ 29. Drs. Kopp and Attard state that the analysis includes beyond-design basis criticality calculations involving fresh fuel assembly misplacements and boron dilution and that none of the scenarios analyzed resulted in criticality in the spent fuel pool. *Id.* Drs. Kopp and Attard's review resulted in their conclusion that Holtec International's calculated criticality values for each region are correct. *Id.* Thus, they conclude that a fuel loading error in the proposed burnup dependent Millstone 3 spent fuel storage racks in Regions 1, 2, and 3 will not cause an inadvertent criticality, or even lead to a significantly increased likelihood of a criticality accident. Kopp/Attard Aff. ¶30. Thus, there is no trade-off and there is no undue risk of a criticality accident related to the proposed amendment.

The second prong of Contention 4 concerns the history of Millstone and alleges that the licensee has a history of being unable to adhere to administrative controls, with respect, *inter alia*, to spent fuel pool configuration. For this second prong, Intervenors' principal reliance is on the December 1997 civil penalty/notice of violation cited by the Board as a basis for admission of Contention 4. LBP-00-2, 51 NRC at 33; *see* Exh. 11. Yet, none of the specific violations cited by the NRC in the December Notice of Violation and Proposed Imposition of Civil Penalties involved spent fuel pool issues at Millstone 3. Cerne Aff. ¶ 6. Also regarding the allegation that Millstone's history reveals an inability to adhere to procedural controls, Mr. Cerne states that the two incidents documented in a 1994 Plant Information Report and in a 1995 Adverse Condition Report produced by NNECO in response to Intervenors' discovery requests are the only two personnel errors documented with respect to Millstone Unit 3 spent fuel movement issues. *Id.* Moreover, both errors were identified during the spent fuel movement process and corrected before any assemblies were physically stored in an incorrect location. *Id.* These incidents occurred prior to the 1996 shutdown and before the initiation of the recovery process that led to a significant Millstone Station cultural change and improvements that permitted the authorization of the restart of Unit 3 by the Commission in June, 1998. *Id.*

Thus, with regard to the administrative controls to be performed in connection with the proposed amendment, there is nothing in Millstone's history that would support a conclusion that Millstone personnel will not be able to perform these actions and Millstone 3's recent history makes such a conclusion even less likely.

James C. Linville, until recently Acting Director of the Millstone Project Directorate in Region 1, has provided an affidavit that focuses on the Millstone plants' recovery from the problems that plagued them in the early 1990s. He reviews the recent history of Millstone, including the Commission's determination to allow Millstone 3 to restart in June 1998 and the improved plant performance since that time that led to the Senior Management's decision after their May, 2000 meeting to remove Millstone 3 from the regional focus Watch List.

As noted above, Mr. Cerne's affidavit is from the point of view of a senior resident inspector; his focus is on the details of the operation of Millstone 3 as they relate to the proposed amendment request. It is Mr. Cerne's expert opinion that there is nothing in Millstone 3's history that would make suspect plant personnel's ability to perform the administrative controls needed in connection with the proposed amendment. Mr. Linville's focus is on improved plant performance, particularly since the plant restarted in June, 1998. He cites improvement in all facets of Licensee performance at Unit 3, including procedure adequacy and adherence, as the basis for Senior Management's decision to remove Unit 3 from regional focus and return it to normal NRC oversight. Linville Aff. ¶ 18.

The focus of Intervenors' discovery was on "errors" at the Millstone plants. In their First Set of Interrogatories and Requests for Production of Documents, filed March 21, 2000, Intervenors requested in their Interrogatory F-1 as follows:

> Please identify all instances of errors (at Millstone or other plants) in managing, moving, placing or tracking fresh or spent fuel and all documents pertinent thereto.

NNECO responded on April 4, 2000, with a list of eleven such "errors," only two of which had occurred at Millstone 3 (Northeast Nuclear Energy Company's Response to CCAM's, CAM's First Set of Interrogatories, April 4, 2000), and on April 20, 2000, produced the eleven documents. Northeast Nuclear Energy Company's Response to CCAM's, CAM's First Set of Document Production Requests, April 20, 2000. The occurrence of two "errors" over the history of the plant, which began operation in 1986, does not establish an inability to perform the administrative procedures connected with the prevention of criticality in the spent fuel pool.

Intervenors were granted a second round of discovery based on their claim that the method NNECO used to identify "errors" was defective, Memorandum and Order (Discovery Rulings, 5/26/00), June 8, 2000, and visited the plant on June 22, 2000, well after the end of the discovery period as originally established by the Board in its Order of February 9, 2000, and as reiterated in its Order of April 19, 2000. Although the Staff has not completed its review of the documents that Intervenors obtained from that visit, some of which were provided by NNECO to the Staff on June 27, 2000, the Staff has not identified anything in the documents it has reviewed that is relevant to Contention 4. As of this writing, the Staff has not yet received and has, therefore, not reviewed, the reactor engineering logs for Millstone Unit 3 RFO-1 through RFO-06 or the 40 procedures that Intervenors requested during the visit to the plant on June 22, 2000. The Staff expects to complete its review of these documents prior to the oral argument scheduled for July 19, 2000.

Intervenors have identified an affiant, James Plumb, who worked at Millstone until January, 1996, but the Staff is not aware of how Mr. Plumb's information might relate to

Contention 4.[18] To the extent that Intervenors rely on information provided by Mr. Plumb concerning his dismissal from Millstone in the downsizing of January 1996, the Board should give little weight to that information.

In the discussion that follows, the Staff provides a context for viewing Mr. Plumb's affidavit.

On June 2, 2000, Intervenors responded to NNECO's Second Set of Interrogatories, dated May 9, 2000. Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone Response to Northeast Nuclear Energy Company's Second Set of Interrogatories, June 2, 2000. The Response indicates that James Plumb, who "worked as a chemistry technician at the Millstone Nuclear Power Generating Station until he was fired in January 1996" (Response to Interrogatory G-4 A.), will provide information regarding his personal knowledge of personnel and operations of the Millstone Nuclear Power Generating Station (Response to Interrogatory G-4 B.). In response to Interrogatory G-4 C., inquiring about the events and allegations he intends to rely upon as a basis for his sworn affidavit and declaration, Intervenors state that Mr. Plumb will rely on the specific events and allegations set forth in the complaint in James Plumb v. Northeast Nuclear Energy Company, CV 96 0537748 (Superior Court, State of Connecticut, Judicial District of New London); the specific events and allegations set forth in the complaint in Arthur J. Roque, Jr., Commissioner, Department of Environmental Protection, V. Northeast Utilities Service Company, et al, CV 97 0575567 (Superior Court, State of Connecticut,

[18] On May 9, 2222, NNECO filed a Second Set of Interrogatories aimed at learning the subject matter of Mr. Plumb's affidavit/statement. This Second Set was said to be occasioned by a letter of June 8, 2000, from Intervenors' counsel to NNECO on which the Staff was not copied.

Judicial District of Hartford); the specific events and allegations set forth in United States of America v. Northeast Nuclear Energy Company, et al, Criminal No.3: 99-CR-211 (RNC), United States District Court, District of Connecticut (Information and Government's Version of the Offenses); the subject matter of OIG Report 99-01s; and prosecutorial activities of the U.S. Department of Justice vis-a-vis current management and staff of the Millstone Nuclear Power Generating Station. In response to Interrogatory G4 D., inquiring about the documents on which Mr. Plumb would rely, Intervenors state that Mr. Plumb will rely on the public records pertaining to the specific events and allegations identified in G4 C.

The Licensing Board should give no weight to the information provided in Mr. Plumb's affidavit. Because Mr. Plumb has not worked at Millstone since January, 1996, his information concerning personnel and operations there is not current information and has no probative value in relation to Contentions 4 and 5, to which it is said to relate.

As regards the complaints on which Intervenors rely, Mr. Plumb's civil suit against the utility for wrongful discharge has been settled. Thus, the complaint is not evidence of anything beyond the mere fact that a complaint was filed; it does not go to the truth of any of the allegations in the complaint.

OIG Report 99-01s, which Intervenors invoke in support of Contention 4, questions, among other things, the Staff's handling of the enforcement action regarding OI Case 1-96-007, which is the report of an OI investigation into the downsizing in which Mr. Plumb was dismissed. This report is not relevant to Contention 4. In any event, a Millstone Independent Review Team (MIRT) was charged by the Commission with the task of reviewing OI Case Nos. 1-96-002, 1-96-007, and 1-97-007, all of which were described or referenced in the Office of the Inspector General (OIG) Event Inquiry, Case No. 99-01S

(December 31, 1998), on which Intervenors rely. (Exhibits 35 ,51). The MIRT issued a report on March 12, 1999, in which it concluded, among other things, that, with regard to Case No. 1-96-007, the case that investigated the downsizing in which Mr. Plumb was dismissed, the available evidence was insufficient to support a conclusion that the three allegers were the subject of discrimination in violation of 10 C.F.R.§ 50.7 (Employee Protection). NRC Millstone Independent Review Team, "Report of Review of Allegations in NRC Office of Investigations Case Nos. 1-96-002, 1-96-007, 1-97-007, and Associated Lessons Learned," Mar. 12, 1999 (Exhibit 35). Thus, the MIRT review did not fault the conclusion reached by OI in Case No. 1-96-007 and the relevance of OIG-99-01s, on. which Intervenors rely, to the matters at issue is not apparent.[19]

Thus, Mr. Plumb's information is not current and it is not relevant to the issues before the Board. Therefore, the Board should give it no weight in making the determinations that the regulations in Subpart K require the Board to make.

Conclusion with respect to Contention 4

Based on the foregoing, there are no genuine and substantial disputes of material fact or of law as to any aspect of Contention 4 and there is no issue raised in Contention 4 that requires the introduction of evidence in an adjudicatory proceeding for resolution.

C. Contention 5

Contention 5, " Significant Increase in Probability of Criticality Accident," as admitted, reads:

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[19] The Staff is unaware of any "prosecutorial activities of the U.S. Department of Justice vis-a-vis current management and staff of the Millstone Nuclear Power Generating Station."

> Will the proposed change in schedule of surveillance of the soluble boron in the fuel pool lead to a significantly increased likelihood of a criticality accident stemming from a misloaded fuel element, during the interval between fuel movements?

Id. at 36. The Board characterized this contention as a factual issue.

1. NNECO's April 17, 2000 Modification of the Proposed Changes to Technical Specifications 3.9.1.2 and 4.9.1.2 Satisfies Intervenors' Concerns Related to Monitoring of Soluble Boron in the Millstone Unit 3 Spent Fuel Pool.

As a part of its March 19, 1999 Application, NNECO proposed to change Millstone 3 Technical Specifications (TS) related to the concentration of soluble boron, as well as surveillance requirements for boron concentration, in the spent fuel pool. TS 3.9.1.2 presently requires that, any time fuel assemblies are in the spent fuel pool, the boron concentration of the pool be maintained at greater than or equal to 1750 parts per million (ppm). Exh. 1, Att. 1, at 6. TS 4.9.1.2 requires that the boron concentration be verified every 72 hours. *Id.* The Application would have amended TS 3.9.1.2 to reflect that the soluble boron concentration be maintained at greater than or equal to 800 ppm during all fuel assembly movements within the spent fuel pool. Ex. 1, Att. 1, at 7. TS 4.9.1.2 would have been amended to require verification of the boron concentration prior to any movement of a fuel assembly within the spent fuel pool, and every 7 days thereafter during fuel movement. *Id.*

Intervenors contend that the proposed TS change increases the probability of a criticality accident in the spent fuel pool at Millstone Unit 3 because it removes the requirement to maintain soluble boron in the pool water at all times. Supplemental Petition at 16, 18. The stated basis for this contention is that, if the requirement is removed, a misloading event could result in a criticality if the misloaded fuel assembly goes undetected and the soluble boron concentration in the pool drops. *Id.* at 18.

On April 17, 2000, NNECO submitted a modification to the proposed TS revisions, which would amend TS 3.9.1.2 to reflect that the soluble boron concentration be maintained at greater than or equal to 800 ppm whenever fuel assemblies are in the spent fuel pool. Exh. 17 at 2. TS 4.9.1.2 would be amended to require verification of the boron concentration every 7 days. *Id.;* Exh. 17, Att. 1, at 1. As discussed below, Intervenors have stated that this boron concentration and surveillance frequency satisfy their concerns underlying Contention 5.

In its First Set of Interrogatories and Requests for Production directed towards Intervenors, dated March 6, 2000, NNECO asked the following:

> Interrogatory No. 5-5: Please state whether the Intervenors have any challenge to the proposed 800 ppm boron concentration with respect to its sufficiency to control criticality, assuming the concentration is verified.

NNECO First Set of Interrogatories, at 7. Intervenors responded, "Petitioners do not challenge the proposed 800 ppm boron concentration, except as described in Contention 6."[20] CCAM/CAM Supplemental Reply to NNECO's First Set of Interrogatories, at 4 (Apr. 25, 2000).

NNECO also asked:

> Interrogatory No. 5-7: Please state the boron surveillance frequency that Intervenors believe would be sufficient to satisfy the concern of this contention.

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[20]Contention 6, discussed below, concerns the legality of taking certain measures pursuant to General Design Criterion (GDC) 62, 10 C.F.R. Part 50, Appendix A. The Board has characterized Contention 6 as a legal issue; it does not concern the science behind the measures.

NNECO First Set of Interrogatories, at 8. In their April 25, 2000 response to this interrogatory, Intervenors stated, in part, "Intervenors would have no objection to a 7-day surveillance frequency."[21] CCAM/CAM Supplemental Response, at 4-5.

Finally, NNECO asked:

> Interrogatory No. 5-8: Please state whether the frequency identified in the response to Interrogatory 5-7 should, in Intervenors' view, be incorporated into Technical Specifications, or whether inclusion in relevant plant operating procedures would be adequate.

NNECO First Set of Interrogatories, at 8. Intervenors responded, "As indicated in the response to Interrogatory No. 5-7, Intervenors believe that the surveillance frequency should be in the Millstone Technical Specifications." CCAM/CAM Supplemental Reply, at 5. NNECO's modified TS proposal incorporates all of these provisions.

Moreover, during the May 10, 2000 deposition of David Lochbaum, an affiant for the Intervenors, the following exchange occurred:

> Q [by Mr. Repka, counsel for NNECO]: Now, are you familiar with the supplemental submittal the company [NNECO] made to revise the proposed tech spec to require surveillance at all times?
>
> A [by Mr. Lochbaum]: The one on April 17, I believe?
>
> Q: I think that is the correct date.
>
> A: Around that date. Yes, I've seen that.
>
> Q: Does that particular proposal resolve your concern on Contention 5?

[21]Intervenors cited to SR 3.7.1.16.1 of NUREG-1431 Rev. 1, "Standard Technical Specifications, Westinghouse Plants" (Apr. 1995), which provides for verification of boron concentration in the spent fuel pool every seven days. CCAM/CAM Supplemental Response to Interrogatories, at 4.

> A: If it is implemented the way it was submitted,[22] it would address my concerns about Contention 5. (. . .)
>
> Q: But if this is implemented in the amendment as issued by the NRC, if this tech spec is incorporated, then you would have, or your Contention 5 would be satisfied?
>
> A: My concerns about Contention 5 would be satisfied, that's correct.

Lochbaum Tr. at 24-26 (Exhibit 31).

Because the Intervenors have demonstrated that the proposal submitted by NNECO to maintain the soluble boron concentration of the spent fuel pool at greater than or equal to 800 ppm, whenever fuel assemblies are in the pool, verifiable every seven days, satisfies the concerns which underlay Contention 5, there is no genuine and substantial dispute of material fact as to Contention 5.

2. CCAM/CAM Cannot Expand the Scope of the Contention as Admitted in this Proceeding.

As stated above, Contention 5 was framed by the Board as a question of fact:

> Will the proposed change in schedule of surveillance of the soluble boron in the fuel pool lead to a significantly increased likelihood of a criticality accident stemming from a misloaded fuel element, during the interval between fuel movements?

The scope of Contention 5 clearly limits the contention to consideration of surveillance of soluble boron in the spent fuel pool.

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[22] During the deposition, Mr. Lochbaum expressed reluctance to withdraw Contention 5, in the event the proposed TS changes were not implemented. See Tr. at 25-26. Although the Staff has not yet issued the license amendment, NNECO continues to demonstrate its commitment to these proposed technical specifications. *See*, *e.g.*, Response to Request for Additional Information, Spent Fuel Pool Rerack (TAC No. MA5137), June 16, 2000 (Exhibit 38)(provided to the Board on June 20, 2000).

Nevertheless, during discovery it became apparent that Intervenors might seek to expand the scope of the contention by now contending the licensee may not take credit for soluble boron when performing criticality safety analyses. See CCAM/CAM Supplemental Reply, at 5 ("Intervenors would have no objection to a 7-day surveillance frequency. However, Intervenors' lack of objection does not constitute an acceptance of credit being taken for soluble boron in either normal or accident conditions.")[23] The assertion of such a theory constitutes an impermissible expansion of the scope of the contention, which should be denied by the Board. As stated by the Appeal Board in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29, 42 (1989), a litigant is not free to:

> modify contentions during an NRC adjudication without cause and observance of the Commission's Rules of Practice. Contentions are simply the issues that define the scope and course of the proceeding. To permit reformulation of contentions every time their proponents file another pleading would be tantamount to rejecting all notions of an orderly and fair administrative process.

The scope of a contention "necessarily hinges upon its terms coupled with its stated bases." *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-98-7, 47 NRC 142, 181 (1998), *citing Public Serv. Co. of New Hampshire* (Seabrook Station, Units 1 & 2), ALAB-947, 33 NRC 299, 372 (1991).

Intervenors did not request reconsideration of the Board's ruling admitting the contention, nor have they sought leave to expand the scope of the contention. Intervenors,

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[23]The double contingency principle, which permits a licensee to take credit for soluble boron in certain scenarios, is discussed below.

therefore, are limited to arguments based upon the contention and bases as admitted by the Board. Any attempt to otherwise expand the scope of the contention is impermissible.

3. Criticality Cannot Occur in the Millstone Unit 3 Spent Fuel Pool Without Two Independent Failures.

The asserted basis for Contention 5 is that NNECO's original proposal to monitor soluble boron only during fuel movements eliminates a barrier against criticality. As discussed in paragraph (2), supra, Intervenors attempt to broaden the contention by implying that the licensee may not take credit for soluble boron in certain abnormal and accident conditions. Credit for soluble boron in certain conditions is permitted by the double contingency principle, which states:

> 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.

Draft Regulatory Guide 1.13, Rev. 2, Draft 1, "Spent Fuel Storage Facility Design Basis" (Dec. 1981), at 1.13-14 (emphasis in original)(Exh. 29).

This principle is the analytical foundation for the Staff's analysis of criticality in spent fuel pools. That principle, as implemented by the Staff, is articulated in 10 C.F.R. § 72.124, which states:

> 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.

This standard, while not directly applicable in this proceeding, is similar to the guidance enunciated in RG 1.13, differing only in that it specifies that "sequential," as well as concurrent, changes be considered in the evaluation.

As discussed above in subsection (1), this asserted basis for Contention 5 is improper. In any event, however, the Staff finds the calculational methods and the assumptions made in NNECO's criticality analysis to be in conformance with NRC guidance. NNECO's criticality analysis was prepared by Holtec International. Exh. 1, Att. 5. The NRC guidelines specify that the maximum effective multiplication factor, k_{eff}, including bias, uncertainties, and calculational statistics, shall be less than or equal to 0.95, with 95 percent probability at the 95 percent confidence level. Grimes letter, Exh. 27, at III-2; Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," (Aug. 19, 1998), at 5 (Exhibit 49). The Holtec International criticality safety evaluation assumed a design limit of 0.945, which is more conservative than the limit specified in the NRC guidelines. Exh. 1, Att. 5, at 4-2.

In addition, the Holtec analysis made several assumptions to ensure that reactivity will be less than calculated reactivity. First, the analysis assumed the moderator in the pool is <u>unborated water</u>, at a temperature within the operating range that results in the highest reactivity. Kopp/Attard Aff. ¶ 24; Holtec Report, Exh.1, Att. 5, at 4-2. No soluble poison (that is, soluble boron), is assumed to be present in the pool under normal operating conditions. *Id.* These assumptions reflect NRC guidelines which specify that the bounding k_{eff} of 0.95 for normal storage be evaluated for the accident condition that assumes the loss of soluble boron. RG-1.13, Exh. 29. The racks were assumed to be fully loaded with the most reactive fuel authorized to be stored in the facility. Kopp/Attard Aff. ¶ 24; Holtec Report, Exh. 1, Att. 5, at 4-2. The analysis neglected neutron absorption by minor structural material (i.e., spacer grids). *Id.* The analysis assumed an infinite array of storage cells, that is, no neutron leakage (except when assessing peripheral effects and certain accidents). *Id.* Finally, in order to maximize the calculated k_{eff}, the analysis incorporated calculational uncertainties and biases, as well as uncertainties due to manufacturing tolerances. *Id.*

Based on an analysis that includes these assumptions, NNECO has determined that a soluble boron concentration of 425 ppm would be sufficient to maintain the required 5 percent subcriticality margin in the event of a fuel assembly misloading event (i.e., a fresh PWR assembly enriched to 5 weight-percent U-235 inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity). Kopp/Attard Aff., ¶ 25. The double contingency principle allows credit for this unlikely event, since only a single independent accident need be considered at one time. RG 1.13, Exh. 29.

Moreover, on June 21, 2000, NNECO submitted, in response to Intervenors' Third Discovery Requests, several calculations performed by Holtec International, for the K_{eff} for each of the three Regions in the pool, assuming various misplacements of fresh fuel assemblies in each Region. This analysis includes beyond-design basis criticality calculations involving fresh fuel assembly misplacements and boron dilution. Kopp/Attard Aff. ¶ 29. None of these scenarios resulted in criticality in the spent fuel pool. *Id.* NNECO has chosen to maintain greater than or equal to 800 ppm of soluble boron in the pool at all times, a demonstrated conservatism. *Id.*, ¶ 25. Consequently, the Staff finds the calculational methods and assumptions made in these analyses to be in conformance with NRC guidelines.

The Staff notes that, as a practical matter, soluble boron in the Millstone Unit 3 spent fuel pool will remain at a level well above the 800 ppm required by technical specifications. TS 3.9.1.1 for Millstone Unit 3 requires a minimum boron concentration of 2600 ppm in all filled portions of the reactor coolant system and the refueling canal during refueling operations. Kopp/Attard Aff. ¶ 27; Millstone Unit 3TS 3.9.1.1 (Exhibit 51). During refueling, the water volume in the spent fuel pool and the refueling canal form a single mass. Kopp /Attard Aff. ¶ 27. As a result, the soluble boron concentration is relatively the same in each of these volumes. *Id.* Therefore, the actual fuel pool boron concentration is approximately 2600 ppm during refueling operations. *Id.* As a practical matter, boron in the spent fuel pool does not disappear after fuel movements, nor is it appreciably diluted over time. *Id.* Any hypothetical event that could dilute the normal boron concentration in the Unit 3 spent fuel pool of approximately 2600 ppm by any significant amount would require such large quantities of water that it would be detectable well before the 800 ppm limit imposed by TS 3.9.1.2 was reached. *Id.*

The proposed change in the schedule of soluble boron surveillance was rescinded by NNECO's letter of April 17, 2000 (Exhibit 17). The staff has determined that the results of the Holtec analyses are correct and concludes that a fuel loading error in the proposed burnup-dependent Millstone 3 spent fuel storage racks in Regions 1, 2, and 3 will not lead to a significantly increased likelihood of a criticality accident.

Conclusion Regarding Contention 5

Based on the foregoing, there are no genuine and substantial disputes of material fact as to Contention 5, and there is no issue raised in Contention 5 that requires the introduction of evidence in an adjudicatory proceeding for resolution.

D. Contention 6

Contention 6, as admitted, reads:

> Does GDC 62 permit a licensee to take credit in criticality calculations for enrichment, burnup, and decay time limits, limits that will ultimately be enforced by administrative controls?

The Board has characterized this contention as a legal issue.

The Intervenors have asserted that reliance on on-going administrative measures to prevent criticality violates GDC 62.[24] Supplemental Petition at 19-20. They enumerate the impermissible measures as maintenance of soluble boron in the pool water and limits on fuel burnup, enrichment and decay time. *Id.* at 20. Intervenors go on to state that, in order to apply GDC 62, "a credible range of accident conditions must be defined" and all possible failures and combination of failures must be analyzed. *Id.* They interpret double contingency principle to required that "the set of non-credible accident scenarios, for the purpose of criticality control, encompasses scenarions involving at least two unlikely, independent and concurrent failures or violations." *Id.* They state, with no support, that the failure of "administrative measures that seek to limit fuel enrichment, fuel burnup or fuel decay time is a likely occurrence," and that such failure will involve more than one fuel

[24] As discussed elsewhere in this presentation, Intervenors have advanced a theory that only on-going administrative measures are prohibited by GDC 62.

assembly. *Id.* at 20-21. As more fully dicussed below, these assertions are without basis and do not raise a substantial and genuine issue of material fact or law.

General Design Criterion (GDC) 62 does not prohibit the use of credit for burnup to maintain subcriticality. GDC 62 provides:

> *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.

10 C.F.R. Part 50, Appendix A, GDC 62.. Nothing in GDC 62 prohibits the use of administrative measures to control the physical systems or processes referenced in that criterion. Similarly, nothing in GDC 62 indicates an intent to prohibit only a certain set of administrative measures - "on-going" measures - or to delineate between ongoing and one time administrative measures. As set forth below, nothing in the history of GDC 62 prohibits the use of such administrative measures, the Staff's consistent practice has been to allow licensees to rely, in part, upon such measures to satisfy that criterion, and the Commission has authorized the use of such controls relating to the prevention of criticality in spent fuel pools

1. <u>Rulemaking history</u>

The Atomic Energy Commission (AEC)[25] added the General Design Criteria to Part 50 in 1971. The AEC went through an extensive process in drafting, redrafting and clarifying the GDC. As a result of these efforts, the AEC staff sent proposed criteria to the

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[25] In 1974, the NRC assumed the AEC's duties with respect to the licensing and regulation of byproduct, source, and special nuclear materials in accordance with the Energy Reorganization Act of 1974. 42 U.S.C. § 5801, et seq. (1994). As used in this written presentation, the "Commission" may refer to either the NRC or the AEC, depending on the time indicated by the context of the discussion.

Commission in a paper dated June 16, 1967. Memorandum from W.B. McCool to Atomic Energy Commission, Exh. 22. On July 11, 1967, the Commission formally published this revised version for comment. 32 Fed. Reg. 10,213, Exh. 23. That version of the proposed GDC included Criterion 66, which provided: "*Prevention of Fuel Storage Criticality.* 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." *Id.* at 10,216. Because the second sentence clearly contemplated the use of means such as geometrically safe configurations and procedural controls to prevent criticality, the statement in the first sentence that "criticality . . . shall be prevented by physical systems or processes" cannot be read to prohibit procedural or administrative controls.

The AEC received numerous comments on this proposed rule making, many of which contained suggestions for changes in the GDC. The AEC received only two comments regarding proposed GDC 66. William B. Cottrell, Director of the Nuclear Safety Information Center at Oak Ridge National Laboratory (ORNL), submitted a comment stating:

> [w]e [do not] 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 as geometrically safe configurations shall be used to insure that criticality cannot occur."

Letter to H.L. Price, AEC, from W.B. Cottrell, NSIC, Exh. 24, at 11. The second comment was received from J.J. Flaherty on behalf of Atomics International. Letter, J.J. Flaherty to Secretary, U.S. Atomic Energy Commission, Exh. 30. This comment suggested that the second sentence of the proposed criterion be replaced with "Inherent means should be used where practicable."

Although there are no available staff documents discussing these comments, it is apparent that the Staff and the Commission did not agree with Oak Ridge that procedural controls should be prohibited, since the AEC did not adopt the suggested language. Rather, the AEC adopted the "preferably by use of geometrically safe configurations" language. *See* Status Report on General Design Criteria, Exh. 25; (Kopp/Attard Aff. ¶ 34; Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969), Exh. 26; Kopp/Attard Aff. 34. Moreover, although the specific word changes suggested in the Atomic International letter were not incorporated in the final criterion, the AEC did incorporate their intent by stating that the use of geometrically safe configurations (inherent means) was the preferred method. The staff again revised the criteria and the Commission adopted them as published in February of 1971.

In promulgating GDC 62, the AEC did not change the part of the draft Criterion 66, as published in 1967, that stated "criticality . . . shall be prevented by physical systems or processes." That language, as set forth above, does not preclude the use of administrative controls. The clause adapted from the second sentence proposed in 1967 and added to that unchanged language, that criticality be prevented "preferably by use of geometrically safe configurations," is, by its own terms, a statement of *preferred* means for preventing criticality. It does not preclude licensees from using administrative controls to aid in satisfying GDC 62.

In addition, GDC 62 applies to fuel handling systems, as well as fuel storage systems. While the fuel handling systems may move only one fuel assembly at a time, administrative controls must be used, for example, to prevent temporary storage of multiple assemblies in close proximity. To adopt Intervenors' reading of GDC 62, *i.e.*, that the GDC does not allow the use of the ongoing administrative controls proposed by NNECO to prevent criticality, would undermine the requirements to prevent criticality applicable to fuel handling, and should be rejected.[26]

This issue was recently addressed in the *Shearon Harris* case. The Licensing Board there held that "there is no clear cut demarcation to differentiate the administrative and non-administrative aspects of criticality control procedures/processes at issue . . . so as to place any of them either inside or outside this label." *Shearon Harris,* LBP-00-12, 51 NRC __, slip op. at 17-18.

The Staff's consistent practice, as approved by the Commission's adjudicatory panels, supports the view that GDC 62 does not prohibit the use of administrative controls, ongoing or otherwise, to prevent criticality.

Nearly every means to prevent criticality, and, in fact, just about every system or process in a plant has some administrative control associated with it, whether it is a

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[26] It is an elementary rule of construction that effect must be given, if possible, to every word, clause and sentence of a statute. Sutherland Stat. Const. § 46.06 (5th Ed). Intervenors' interpretation of GDC 62 would lead to anomalous results as applied to fuel handling systems, thus illustrating the wisdom of interpreting provisions as a whole. As set forth in Sutherland:

> It is always an unsafe way of construing a statute or contract to divide it by a process of etymological dissection, and to separate words and then apply to each, thus separated from its context, some particular definition given by lexicographers and then reconstruct the instrument upon the basis of these definitions. An instrument must always be construed as a whole, and the particular meaning to be attached to any word or phrase is usually to be ascribed from the context, the nature of the subject matter treated of, and the purpose or intention of the parties who executed the contract or of the body which enacted or framed the statute or constitution.

Sutherland Stat. Const., § 46.05 (5th Ed).

surveillance, limiting condition of operation, testing, or some other administrative control. The Intervenors object to the use of credit for enrichment, burnup or decay time as verification in selecting the placement of spent fuel assemblies in the SFP as an on-going administrative control. However, because human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. Kopp/Attard Aff. ¶ 15. Moreover, the enrichment, burnup and decay of the fuel are themselves physical processes. *Id.*, ¶ 36, 37. In addition, the Staff has been authorizing the use of credit for burnup in selecting fuel assemblies for locations in spent fuel racks for eighteen years or more. *See id.*, ¶ 39. There has never been a report of a criticality accident in any spent fuel pool. *Id.*, ¶ 15, 40. Moreover, NNECO has been utilizing credit for burnup in the Millstone 3 SFP for several years with no adverse consequences.

In addition, the Staff has approved administrative controls to prevent criticality in spent fuel pools in amendments litigated before the Commission's adjudicatory panels. *See, e.g., Consumers Power Co.* (Big Rock Point Nuclear Plant), ALAB-725, 17 NRC 562, (1983). *Big Rock Point* dealt with the use of a makeup line, a physical system, to maintain water level in the spent fuel pool at that facility. *Id.* at 571. In its decision, the Appeal Board identified the makeup line as "remotely controlled." *Id.* at 564-65, 571. Such remote control would rely on administrative controls so that plant operators could exercise control over this system.

Moreover, at least one Atomic Safety and Licensing Board has accepted administrative controls to control the placement of fuel assemblies in spent fuel pools. *See*

Florida Power & Light Co. (St. Lucie Nuclear Power Plant, Unit 1), LBP-89-12, 29 NRC 441 (1989), *aff'd on other grounds*, ALAB-921, 30 NRC 177 (1989). The intervenor in that proceeding raised the following criticality[27] contention regarding misplacement of a fuel assembly:

> The mechanisms which prevent the erroneous insertion of a fuel assembly into a storage cell such that the prescription of Standard Review Plan ("SRP") Section 9.1.2, Part III, 2.b., that it is not possible for a "fuel assembly . . . (to) be inserted anywhere other than a design location," have not been demonstrated.

St. Lucie, LBP-89-12, 29 NRC at 454. The spent fuel pool was divided into two regions. Only fuel assemblies that had reached the required burnup could be stored in Region 2; but it was possible to "insert an assembly with less than the requisite burnup in Region 2." *Id.* at 455. The *St. Lucie* Licensing Board referenced Staff guidance, which allowed for administrative controls, with written procedures to prevent misplacement, and described Florida Power and Light's (FPL) administrative controls to assure proper placement of fuel assemblies. *Id.* The *St. Lucie* Licensing Board held that:

> the foregoing procedures and restraint used in the handling of fuel assemblies in the spent fuel pool are adequate to provide reasonable assurance that fuel will be stored in the prescribed areas of the pool. The procedures satisfy the guidelines of SRP 9.1.2 and will ensure against improper storage of fuel assemblies.

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[27] *See Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 1), LBP-88-27, 28 NRC 455, 473-75 (1988).

Id. at 456. Clearly, the Board in *St. Lucie* recognized that administrative controls are permissible to control criticality in a spent fuel pool.[28] *See also Shearon Harris*, LBP-00-12, 51 NRC ____.

2. The Commission has authorized the use of administrative controls relating to the prevention of criticality in spent fuel pools

In 1998, the Commission issued a final rule on criticality monitoring requirements in Part 50. 10 C.F.R. § 50.68. Section 50.68 provides that licensees may elect to comply with the criteria in that regulation, rather than choosing to comply with 10 C.F.R. § 70.24, which requires the use of a criticality monitoring system. In proposing the rule, the Commission stated:

> The [NRC] is amending its regulations to provide light-water nuclear power reactor licensees with greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area where the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

Proposed rule, Criticality Accident Requirements, 62 Fed. Reg. 63,911 (1997). The final rule included a similar statement. Final rule, Criticality Accident Requirements, 63 Fed. Reg 63,127 (1998). Responses to comments in the notice of issuance of the final rule

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[28] Other proceedings have involved the application of GDC 62. *See e.g., Florida Power and Light Co.* (Turkey Point Plant, Units 3 & 4), Nos. 50-250-OLA-2; 50-251-OLA-2, 1999 NRC LEXIS 13381, at *13396-98 (March 25, 1987)(unpublished)(use of burnup). While this proceeding did not involve any dispute over the meaning of GDC 62, it is illustrative of the Staff's practice regarding the use of administrative controls to prevent criticality in spent fuel pools. It involved precisely the same means for controlling criticality at issue here: credit for burnup. *Id.*

explicitly demonstrate that the Commission was aware of licensee's use of administrative controls to prevent criticality in spent fuel pools. *Id.* at 63,128.

Section 50.68(b) specifies eight criteria. The criteria in Sections 50.68(b)(2), (3), and (4) discuss credit for soluble boron in the fuel pool water. Section 50.68(b)(2) provides:

> The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks 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. The evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

10 C.F.R. § 50.68(b)(2). In establishing this criterion, the Commission clearly approved the use of administrative means to prevent boron dilution events (flooding with unborated water) to prevent criticality. *See also* 10 C.F.R. § 50.68(b)(3).

Similarly, 10 C.F.R. § 50.68(b)(4) addresses credit for soluble boron relating to spent fuel storage racks, and reads:

> 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 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.

As this regulation indicates, the Commission is aware of and has approved the use of administrative controls in conjunction with physical systems and processes as means of preventing criticality in spent fuel pools. Likewise, the use of administrative controls to prevent flooding with unborated water and optimum moderation of fresh fuel has also been approved. These methods would fit into the Intervenors' definition of "on-going" administrative measures, yet, as noted above, they have been approved by theCommission. Since the Commission has approved the use of administrative controls to control boron concentration to prevent criticality in spent fuel pools, then it is clear that the Commission does not believe that the use of administrative controls, whether one time or on-going, in conjunction with physical controls violates GDC 62.

3. There is no Basis for the Theory That GDC 62 Prohibits the Use of Ongoing Administrative Controls

As discussed elsewhere in this presentation, Intervenors offer a theory which assumes that there are two kinds of administrative measures - one time, which are acceptable under GDC 62, and ongoing, which are unacceptable. They provide no regulatory, statutory, or scientific support for this theory, just the untested, unsupported, opinion of their witness, Gordon Thompson. There is absolutely no support for this distinction in the applicable Commission's regulations or the case law. The Commission has never proposed or accepted such a distinction relating to spent fuel pools, fuel handling or reactor operation. In fact, as demonstrated above, the Commission has previously approved such measures. There is nothing in the rulemaking history of GDC 62 or in GDC 62 itself which supports this distinction. In fact, the distinction was rejected by the licensing board in *Shearon Harris*, which found no basis for distinguishing administrative controls as requested by the intervenor therein. *Shearon Harris*, LBP-00-12, 51 NRC __, slip op. at 17-18.

The Intervenors have added nothing new to his discussion in this case. For example, when asked in the Staff's interrogatories to provide the basis for their position that administrative measures are not permitted by GDC 62, the Intervenors answered by referencing Orange County's filing of January 4, 2000, in the Shearon Harris case, at pages 18-37. Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone's Reply to NRC Staff's First Set of Interrogatories, at 9, April 8, 2000. Therefore, since Intervenors have not, and likely will not, supply any further support for its position other than Dr. Thompson's unsupported theory, the Staff submits that the theory should be similarly rejected by this Board as not based in logic, science, regulation or law, or on competent expert opinion.

4.. The use of credit for burnup is permissible. Criticality will not occur in the Millstone 3 spent fuel pool without two independent failures, as specified in draft RG 1.13.

Although not directly relevant to the issue of whether GDC 62 permits the use of administrative controls as an aid to prevent criticality, the Intervenors have made an argument regarding the double contingency principle specified in draft Reg. Guide 1.13 in this context. As discussed in the Staff's argument related to Contention 5, the double contingency principle will not be violated by the proposed criticality prevention means. As has been amply demonstrated in the applicant's submittal and analyses, and as reviewed by the Staff, the misplacement of one fuel assembly will not cause the 5% subcriticality margin to be violated, assuming the spent fuel pool water contains 425 ppm of boron (which is less than the 800 ppm of boron maintained in the pool water).[29] Exhibit 1, Attachment 5 ; Kopp/Attard Aff. ¶ 25. As has been further demonstrated in the licensee's analysis, two failures, that is, misplacement of one fuel assembly AND loss of all boron, will not cause

[29] While the misplacement of a fuel assembly may be considered to be one failure, not discovering the misplacement would be considered a second failure.

criticality. Exh. 1, Att. 5. In fact, as has been further demonstrated, misloading of multiple fresh fuel assemblies will not cause criticality. Kopp/Attard Aff. ¶ 29.

As set forth above, the Intervenors are not capable of performing criticality analyses,. (Exh. 15) The Staff has reviewed NNECO's analyses and has found them to be acceptable. Kopp/Attard Aff. ¶ 30. Therefore, these analyses are unchallenged and there is no issue as to their accuracy or acceptability. Since the Licensee has demonstrated that criticality will not be reached even under highly improbable beyond design basis events - the Intervenors' assertion must be rejected.

The Intervenors assert that failure of administrative measures is a likely event. To support this assertion, the Staff believes that they may produce several reported incidents of alleged misloadings. The Staff notes the following about these incidents. Only a handful relate to misloading of spent fuel in spent fuel pools,[30] and none of them resulted in criticality in a spent fuel pool. Moreover, Intervenors cannot demonstrate that such events are likely at Millstone 3.

The Intervenors state that the double contingency principle requires an analysis of all the possible sets of scenarios that may cause criticality. Supplemental Petition at 20. This would require an identification of all combinations of events and then an analysis of whether they were unlikely and independent. The Staff submits that such an analysis is not required. The analysis done by NNECO in this case meets the double contingency

[30] When viewed in relation to the number of SFPs in which credit for burnup is permitted (50 or more), the number of fuel movements per year (one third of each core is changed every 18 months to two years) and the number of years involved (credit for burnup has been permitted since the early 1980s), it is clear that such misplacements are extremely rare and unlikely. To the extent that the Intervenors now seek to define misplacements to include misalignments in the reactor core, that should be rejected by the Board as irrelevant and immaterial.

principle. NNECO's analysis identifies the worst unlikely independent single events (*e.g.* loss of boron) and evaluates whether they result in criticality. If they do not reach criticality and another unlikely independent event is required before criticality is reached, then the double contingency principle is met. So, in accident analysis, if one assumes a fuel assembly misplacement, then one can assume presence of boron. If one assumes loss of boron, one does not have to also assume fuel misplacement. Exhibit 49 at 4.

Conclusion with respect to Contention 6

Based on the foregoing, there are no genuine and substantial disputes of material fact or of law as to any aspect of Contention 6 and there is no issue raised in Contention 6 that requires the introduction of evidence in an adjudicatory proceeding for resolution.

NRC STAFF RESPONSES TO BOARD'S QUESTIONS FOR PARTIES

On May 23, 2000, the Atomic Safety and Licensing Board (Board) issued a memorandum requesting that the parties address several questions relevant to the three admitted contentions in this proceeding. The NRC Staff hereby responds to the questions posed therein.

A. Contention 4 (fuel assembly misplacement)

> 1. Is there any statistical basis for asserting that the likelihood of fuel assembly misplacement is either high or low? If so, what is the basis and what are the statistics? Are this basis and the accompanying statistics applicable to the Millstone-3 reactor or must other factors be taken into account? If there is no such basis, how do you decide on the likelihood of fuel assembly misplacement? (Intervenors, Staff)

STAFF RESPONSE:

No. The Staff is not aware of any statistical study regarding the likelihood of fuel assembly misplacement. Nonetheless, although a quantitative statistical basis has not been established, the number of reported fuel misloading events in spent fuel pools is small in comparison to the large number of fuel movements each year. This indicates that the likelihood of a misloading event is relatively small. There have been reports of fuel misloading events over the years, but many of them are not applicable to spent fuel pool storage since they involved reactor core misloadings or misorientations. Of those applicable to spent fuel storage, most were detected and corrected within a short period of time because of independent checks required by plant procedures, indicating that administrative controls are effective in precluding permanent fuel misloadings. None resulted in any inadvertent criticality events. The Staff has not developed any statistics on the likelihood of a fuel assembly misplacement in a spent fuel pool; however, the Staff considers fuel misplacement in spent fuel storage racks, including those at Millstone Unit 3, to be an unlikely event for the following reasons.

First, proposed TS 3.9.13 will control fuel storage limitations and selection procedure SP 31022, Rev. 4, Exhibit 8, described above, will control fuel assembly selection. Therefore, both TS as well as plant procedures would have to be violated for a fuel assembly misplacement to occur. In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. Finally, the burnup limit curves (Figures 3.9-1, 3.9-3, and 3.9-4) proposed for the Millstone Unit 3 TS for safe storage in regions 1, 2, and 3, are based on minimum burnup requirements. These are bounding values that result in just meeting the 5% subcriticality margin for storage pools in region 1, 2, and 3.

In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). Such fuel assemblies, therefore, should fall into the acceptable burnup domain specified in these figures, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. Consequently, the placement of a fuel assembly that does not meet the technical specification burnup requirements into a large storage pool, such as the Millstone 3 SFP, and the continued failure to detect this misplacement, is a highly unlikely event at Millstone 3. Multiple misplacements are even more unlikely. It is highly unlikely that a single failure in the administrative controls or the management process will lead to misplacement or multiple misplacements. Such multiple misplacements, with or without boron dilution, leading to criticality, are highly improbable and well beyond the application of the double contingency principle. Although there have been several reported fuel assembly misplacements in spent fuel pools at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in detecting and correcting fuel misloadings.

In the absence of a statistical database to ascertain the consequences of a misplaced assembly, the licensee performs a calculation that analyzes a fuel misloading event involving the misplacement of the most reactive fuel assembly, regardless of the probability of its occurrence. NRC guidance (specifically, the April 14, 1978 letter from B.K. Grimes to all power reactor licensees, at 1.13-9) requires such an analysis to demonstrate compliance with both the Staff's 5 percent subcriticality margin criterion and the subcriticality requirement of General Design Criterion (GDC) 62.

2. How are fuel assemblies identified initially and during subsequent fuel element transfers? (Licensee)

The Staff has no comment on question A(2).

3. If there is a difference between two observers as to the identity of a fuel assembly, how is that difference resolved? (Licensee)

The Staff has no comment on question A(3).

4. When assigning burn-up plus decay time administrative controls, how are the dividing lines between fuel assemblies of various burn-ups decided, and how are uncertainties in burn-up treated? (Licensee, Staff)

STAFF RESPONSE:

The dividing line between fuel assemblies of various burnups is decided based on storage rack composition and configuration. First, the burnup required to allow fuel of the maximum initial enrichment used in the core (typically 5 weight percent U-235) to be safely stored in the racks and meet the 5 percent subcriticality margin is determined. Similar calculations are made for fuel assemblies of lower initial enrichment requiring less burnup to meet the 5 percent margin.

Decay time credit is an extension of this process and includes the time an assembly has been discharged from the core and residing in the pool as a variable. Spent fuel decay time credit results from the radioactive decay of fissile isotopes in the spent fuel to daughter isotopes, which are non-productive neutron absorbers. Thus, the additional decay while cooling in the spent fuel pool further reduces the spent fuel's reactivity. This loss in reactivity due to decay allows a reduction in the minimum burnup needed to meet the reactivity requirements and typically is applicable to older fuel that has been stored in the racks for a period of years.

A large part of the uncertainty in burnup calculations derives from uncertainties in fuel and moderator temperatures and the spectral effect of soluble boron during core operation. Bounding values of these operating parameters are assumed in order to assure the highest plutonium production and, consequently, conservatively high values of reactivity during burnup. The specific code used for burnup calculations is then benchmarked against various critical experiments (including plutonium-bearing fuel), other codes, and reactor operations, to determine calculational bias and uncertainty. In addition, a burnup-dependent uncertainty in reactivity is statistically applied (square root of the sum of the squares) in the determination of the acceptable burnup versus enrichment combinations. Finally, since uniform axial burnup is typically assumed, a correction to account for actual axial burnup distribution is added, if necessary, to the maximum calculated $k_{eff.}$ Thus, all the uncertainties associated with burnup predictions are factored into the calculated k_{eff} to assure that the final k_{eff} is conservative.

> 5. Can it be stated that unirradiated fuel planned for use in the Millstone-3 plant has the highest level of reactivity worth as compared with Millstone-3 irradiated fuel? If not, what level of fuel burnup or combination of time of decay and burnup provides the highest level of reactivity worth? (Licensee)

The Staff has no comment on question A(5).

> 6. Please describe the difference between Wt& U-235 and W/o U-235 (% weight by volume). What, if any, significance is there to any differences? (Licensee)

The Staff has no comment on question A(6).

7. With respect to the determination of storage placement of fuel assemblies, can it be said that Figures 4.1.1, 4.1.2, and 4.1.3 of the non-proprietary version of the Licensee's license-amendment application, dated March 19, 1999, are the principal vehicles by which placement determination is made? If not, what other considerations are there? (Licensee)

The Staff has no comment on question A(7).

2. Contention 5 (borated water)

a. Please describe the chemical form and/or compound used to create and maintain soluble boron concentrations in the spent fuel pool. Please provide information as to its solubility over the range of possible conditions, particularly temperature, in the spent fuel pool. Are there any volumes in the pool cold enough to cause the boron to come out of solution? (Licensee)

The Staff has no comment on question B(1).

b. How is the boron concentration measured? How accurate is the test for boron concentration? (Licensee)

The Staff has no comment on question B(2).

c. How frequent a check on boron concentration is needed to give adequate assurance of boron concentration? What determines the needed frequency? (All parties)

STAFF RESPONSE:

The amount of soluble boron in spent fuel pool water is generally determined by the amount of boron required in the reactor core by technical specifications to maintain at least a 5 percent subcriticality margin in the core during refueling. During refueling operations, the water volume in the reactor vessel, the refueling cavity, the refueling canal, and the spent fuel pool form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes. Once the proper amount of soluble boron is installed in the spent fuel pool water, it is very difficult to dilute it. In addition, there are high and low water level alarms in the fuel handling building and in the control room, which, if water were added or lost, would alert the operators to an increasing or decreasing water volume in the pool and, thus, a possible flooding or loss of shielding event. Increasing or decreasing water levels are also readily detectable visually by plant staff in the spent fuel pool area. Any postulated accident condition that could significantly dilute the pool boron concentration would require such large quantities of water (hundreds of thousands of gallons) flowing over the pool and into other areas such as stairwells, that it would be readily detectable long before the soluble boron concentration could be reduced to a level that has any significance with regard to criticality.

Therefore, while there is no specific prescribed frequency for checking soluble boron concentration, based upon the foregoing, the weekly periodic sampling of soluble boron concentration proposed by NNECO is adequate to ensure that any credible dilution accident would be detected and corrected on a timely basis. Finally, although any postulated event that could dilute an appreciable amount of soluble boron is considered to be highly unlikely, a complete loss of all soluble boron was analyzed for the Millstone Unit 3 spent fuel pool and shown to maintain the staff's 5 percent subcriticality margin criterion as well as the GDC 62 requirement for subcriticality.

d. During the time when boron concentrations have been measured every 72 hours, what is the largest change observed in boron concentration? (Licensee)

The Staff has no comment on question B(4).

e. What are the record-keeping and reporting requirements with respect to boron surveillance and its concentration? (Licensee)

The Staff has no comment on question B(5).

c. Contention 6 (GDC 62, etc.):

a. Referring to 10 C.F.R. § 50.68(b), what is your definition of reactivity? (Relate this to standard textbook definitions in, e.g., Hetrick, "Dynamics of Nuclear Reactors," Glasstone & Edlund, "The Elements of Nuclear Reactor Theory," etc.) (All parties)

STAFF RESPONSE:

The definitions used in Hetrick, "Dynamics of Nuclear Reactors," and Glasstone & Edlund, "The Elements of Nuclear Reactor Theory," is the same as that given in most standard nuclear engineering textbooks, including Lamarsh, "Introduction to Nuclear Engineering," 2nd Edition, at 282, that is: (k-effective -1) / (k-effective). This definition is applied to a system. The definition of this system, in turn, varies depending on the problem of interest. For example, a system might be an individual fuel assembly immersed in pure water or multiple depleted assemblies stored in a poisoned rack with water as moderator. It has been common practice in the nuclear industry to use the phrase "fuel assembly reactivity" to refer to the reactivity worth of a fuel assembly.

b. What is the meaning of the phrase "maximum fuel assembly reactivity" used in 10 C.F.R. § 50.68(b)(4)? How is the maximum fuel assembly reactivity measured? (All parties)

STAFF RESPONSE:

Based on comments received after publication of the proposed 10 C.F.R. § 50.68 in the Federal Register, the Commission replaced the phrase "maximum permissible U-235 enrichment" with "maximum fuel assembly reactivity" since most boiling water reactors (BWRs) have fuel rods of varying enrichments within a bundle and BWR technical specifications are defined in terms of the infinite multiplication factor of an assembly in standard core geometry rather than enrichment. The phrase is also appropriate for pressurized water reactors (PWRs) since enrichment (the percent of U-235 present in the fuel by weight) decreases with fuel burnup. The specification of a maximum enrichment value is only appropriate for fresh fuel. The maximum fuel assembly reactivity is not a measurable quantity but is calculated based on fuel assembly design, initial enrichment, irradiation history in the core, and decay time. It refers to the fuel assembly configuration and nuclide composition that results in the highest k-effective when placed in the spent fuel storage racks.

c. When was credit for burn-up first considered in spent fuel pools? Were the considerations involved discussed with the ACRS or the Commission? (Staff)

STAFF RESPONSE:

The NRC first approved burnup credit in spent fuel pool storage criticality safety analyses in the early 1980s. Plants that were initially approved for burnup credit include Fort Calhoun (1983), St. Lucie 2 (1984), Ginna (1984), Turkey Point 3 & 4 (1984), and Summer (1984). It is not known whether the considerations involved were discussed with the Advisory Committee on Reactor Safeguards(ACRS) or the Commission. A memorandum was sent from G. Arlotto to R. Fraley (ACRS) on September 23, 1981, attaching Draft 1 of Regulatory Guide 1.13, Revision 2, for review by the ACRS Regulatory Activities Subcommittee (Exhibit 49). This draft included sections allowing credit for burnup in spent fuel pool storage. However, since the revision was never issued, it is unclear whether it was reviewed by the ACRS.

d. Inasmuch as current spent fuel pool practices appear to have been followed at the time 10 C.F.R. § 50.68 was first formulated, why did the proposed rule or the Statement of Considerations for the final rule not contain an explicit discussion of administrative controls on burn-up and decay time? Why was 10 C.F.R. Part 50, Appendix A, Criterion 62 (GDC 62) (which has been in effect since 1971) not amended or clarified at the time 10 C.F.R. § 50.68 was adopted (during 1998), to reflect that administrative controls on burn-up and decay time (as well as factors explicitly mentioned in GDC 62) could be considered? (Staff)

<u>STAFF RESPONSE:</u>

10 C.F.R. § 50.68 was formulated to give licensees the option of either meeting the criticality accident monitoring requirements of 10 C.F.R. § 70.24 in handling and storage areas for special nuclear material and including criticality accident radiation monitors which would indicate a criticality accident had occurred, or complying with the requirements stated in 10 C.F.R.§ 50.68, which preclude criticality through procedures and design. It was intended primarily for criticality prevention of fresh fuel arrays in handling and temporary storage areas before their storage in fixed geometry racks that are designed to prevent criticality. Therefore, the rule simply states the staff's 5 percent subcriticality criterion for designed spent fuel pool storage racks and does not elaborate on the various acceptable ways to meet the criterion.

The Commission did not need to amend GDC 62 to show that administrative controls could be used since, as written, it does not say that they cannot be used. The GDCs are general, minimum requirements governing nuclear power plant design and operations. Guidance on how to meet these requirements has been developed over the years by the Staff, primarily through standard review plans (SRPs) and other Staff guidance, such as the Grimes letter (Exhibit 27) and the Memorandum from L.I. Kopp, NRC, to T. Collins, NRC, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," U.S. Nuclear Regulatory Commission (August 1998) (Exhibit 48). Although the staff is not aware of any plans to amend the GDCs, there are plans to update the SRPs. Updates to the fuel storage SRPs will reflect the fact that credit for burnup and decay time and the use of administrative procedures to determine fuel assembly selection and placement are acceptable means for meeting the GDC requirements.

e. What is the scope of the phrase "physical systems or processes" as used in GDC 62? (All parties)

STAFF RESPONSE:

The term "physical systems or processes" refers to the methods available and used to prevent inadvertent criticality in spent fuel pool storage. These systems or processes include geometric separation of assemblies, solid and soluble neutron absorbers, the use of storage racks, and fuel reactivity. Fuel reactivity is determined by fuel assembly design and composition, initial enrichment, and burnup. Fuel burnup, a natural outcome of reactor operation, is also a physical process.

f. Are "procedural controls" included in the scope of "physical processes?" (All parties)

STAFF RESPONSE:

Yes. Implementation of each of the physical processes described above requires some form of administrative controls. For example, administrative controls are used in the fabrication of the storage racks to assure proper geometric spacing and solid neutron absorber placement. Likewise, administrative controls are used when placing fuel assemblies in the storage racks based on accumulated burnup, in order to verify that the required burnup has been attained and that the assemblies are placed in the designated locations.

g. If a licensee changes from an 18-month to a 24-month fuel cycle, what changes must or should the licensee make in the spent-fuel pool? (Intervenors, Staff)

STAFF RESPONSE:

Unless the new, longer operating cycle requires higher enriched fuel than used previously at the plant, no changes are required. If an enrichment increase is made, a new criticality safety analysis is required for the spent fuel storage pool (and fresh fuel storage racks) to confirm that the Staff's 5 percent subcriticality criterion and GDC 62 requirements are met. If the longer cycle results in higher fuel assembly burnup, the discharged fuel will be lower in reactivity than previously discharged fuel and will meet existing Millstone 3 Technical Specification minimum burnup limitations on fuel storage.

CONCLUSION

Based on the foregoing, there are no genuine and substantial disputes of material fact as to any aspect of Contentions 4, 5 or 6 and there is no issue raised in any of the contentions that requires the introduction of evidence in an adjudicatory proceeding for resolution.

Respectfully submitted,

Ann P. Hodgdon
Counsel for NRC staff

Susan L. Uttal
Counsel for NRC staff

Brooke D. Poole
Counsel for NRC staff

Dated at Rockville MD
this 30th day of June, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Docket No. 50-423-LA-3
NORTHEAST NUCLEAR ENERGY	)	
COMPANY	)	ASLBP No. 00-771-01-LA
	)	
(Millstone Nuclear Power Station, Unit 3)	)	
	)	

AFFIDAVIT OF LAURENCE I. KOPP AND ANTHONY C. ATTARD IN SUPPORT OF THE NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON TECHNICAL CONTENTIONS 4, 5 AND 6

Laurence I. Kopp and Anthony C. Attard, being duly sworn, do hereby state as follows:

1. I, Laurence I. Kopp, have been employed by the U.S. Nuclear Regulatory Commission (NRC), and its predecessor, the Atomic Energy Commission (AEC), since 1965. My current position is Senior Reactor Engineer in the Reactor Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation (NRR). My responsibilities include review and evaluation of the criticality aspects of on-site fuel storage at commercial nuclear power reactors. I have a Ph.D. degree in Nuclear Engineering from the University of Maryland, a Master of Science degree in Physics from Stevens Institute of Technology, and a Bachelor of Science degree in Physics from Fairleigh Dickinson University. I have 42 years experience in the nuclear power industry, including 5 years at the Martin-Marrietta Nuclear Division and 2 years at the Westinghouse Astronuclear Division. Exhibit 20 is a statement of my professional qualifications.

2. I, Anthony C. Attard, have been employed as a reactor Physicist/Engineer by the U.S. Nuclear Regulatory Commission (NRC) for almost 10 years. My tenure at the NRC has been spent exclusively in the Reactor Systems Branch (SRXB). My assignments cover a wide range of licensing and operating reactor issues, such as, reloads, technical specification changes, accident analysis, advanced reactors, boron dilution transients, probabilistic risk assessment methods. I have a Ph.D. in Nuclear Physics and Engineering from Carnegie-Mellon University and a Bachelor of Science degree in Mathematics and Physics from the University of Michigan. I have 25 years experience in the nuclear power industry, commercial and military reactors. Exhibit 21 is a statement of my professional qualifications.

3. The purpose of this affidavit is to address the Connecticut Coalition Against Millstone (CCAM) and the Long Island Coalition Against Millstone (CAM) Contentions 4, 5 and 6 as set forth in the Atomic Safety and Licensing Board (Board) Prehearing Conference Order (Granting Request for Hearing) of February 9, 2000. *Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 3), LBP-00-02, 51 NRC 25 (2000).

4. In a letter from R.P. Necci to the NRC, dated March 19, 1999 Northeast Nuclear Energy Company (NNECO or Licensee) submitted a request to rerack the spent fuel pool at the Millstone Nuclear Power Station, Unit No. 3. Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification, Spent fuel Pool Rerack (TSCR 3-22-98)(Application) (Exhibit 1).

5. In preparation for this affidavit, we reviewed the criticality aspects of the NNECO application for the proposed license amendment as well as the correspondence and technical documents identified below.

6. Criticality is the achievement of a self-sustaining nuclear chain reaction. The chain reaction proceeds as atoms of a fissile material absorb slow (thermal) neutrons and split (fission) into new lighter atoms (i.e., fission products) and additional neutrons that, in turn, interact with additional fissile atoms. Neutrons resulting from fission have high energy and are called "fast" neutrons. Fast neutrons are not readily captured in U-235, the fissile material originally present in fresh fuel. Rather, a neutron must lose energy and "slow down," or become "thermalized" (a thermal neutron), in order to be readily captured in U-235 and cause fission.

7. In order for fast neutrons to slow down, they must collide with, and transfer energy to, atoms. This process is called "moderation." A light element (such as hydrogen) is an effective moderator because the mass of its nucleus is on the same order as that of a neutron. Therefore, upon initial collision, the neutron imparts most of its energy to the hydrogen nucleus and becomes thermalized. Water, with its high hydrogen content, is the moderator in a light water reactor (LWR) such as Millstone Unit 3.

8. After being created through fission, during the process of moderation, and after reaching thermal energy levels, a neutron may undergo several events. It may be absorbed by nonproductive capture in the fuel, the moderator, or the structural materials. It may leak from the reactor system and either be reflected back into the system or be lost. Finally, it may be absorbed by the U-235, cause fission, and produce more fast neutrons.

9. When the process continues on its own, the system of atoms of fissile material is said to be critical. The measure of criticality is the effective neutron multiplication factor, k-effective, or k_{eff}. The multiplication factor is the ratio of the rate of neutron production to neutron loss due to fission, nonproductive capture and leakage. Well-developed mathematical models (equations) exist in present-day computer codes and are used to compute k_{eff}. Criticality is achieved when k_{eff} is equal to 1.0. When k_{eff} is less than 1.0, the system is subcritical. When k_{eff} is greater than 1.0, the system is supercritical. Criticality can only occur in an array of LWR fuel if sufficient fissile material is available in a near-optimum geometry and a moderator (water) is present. As previously mentioned, no array of LWR fuel can achieve criticality without water moderation present in the array.

10. "Reactivity" is defined as $(k_{eff} - 1)/k_{eff}$. When fuel is irradiated in a reactor as a result of operation and power generation, the reactivity of the fuel decreases. This reduction of reactivity with irradiation is called "burnup." Burnup is caused by the change in fissile content of the fuel (i.e., depletion of U-235 and production of Pu-239 and other fissile actinides), the production of actinide neutron absorbers, and the production of fission product neutron absorbers. A reload analysis is performed before each reactor operating cycle in which the burnup of each fuel assembly during the cycle is predicted. These calculations are confirmed during the cycle by measurements of various operating characteristics, such as boron concentration and power distribution. After every operating

cycle (typically 1 to 2 years), approximately 1/3 of the fuel in a reactor is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment. This irradiated (or spent) fuel is generally placed in a spent fuel pool at the reactor site and is replaced in the reactor by fresh (unirradiated) fuel.

11. Contention 4 states:

> 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.

12. Our response to Contention 4 is contained in the following paragraphs.

13. Licensees have used administrative procedures in essentially all burnup-dependent storage pools since the early 1980's. These procedures generally consist of verification that the licensee has selected a fuel assembly that has zero burnup (new fuel), or assemblies that have achieved the required amount of burnup, based on plant operating records, and the licensee has stored it in the intended position in the spent fuel pool. Administrative procedures are simply mechanisms for verifying physical processes and implementing physical controls. Section 4.2.1 of American National Standards Institute (ANSI) standard ANSI/ANS-8.1-1983, states that nuclear criticality safety may be achieved by controlling one or more parameters of the system within subcritical limits and that control may be exercised administratively through procedures. ANSI, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, Oct. 1983. The NRC endorsed ANSI/ANS-8.1.1983 in revision 2 to Regulatory Guide 3.4. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.4, Revision 2, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuel and Materials Facilities," Mar. 1986 (Exhibit 48).

14. In addition, the Code of Federal Regulations, 10 C.F.R. § 50.68, allows the use of administrative controls to prevent inadvertent criticality in fuel handling and storage. The

Commission developed 10 C.F.R. § 50.68 to allow holders of a construction permit or operating license for a nuclear power reactor issued under 10 C.F.R. Part 50 relief from the 10 C.F.R. § 70.24 requirement to maintain a criticality accident monitoring system in each area where nuclear fuel is handled, used, or stored, if criticality is precluded in these areas. Specifically, 10 C.F.R. § 50.68(b)(1) allows a licensee to rely upon plant procedures to "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 conditions feasible by unborated water." Section (b)(2) and (b)(3) of 10 C.F.R.§ 50.68 allow licensees to use administrative controls or design features or both to prevent accidental flooding of new fuel racks to preclude criticality.

15. Therefore, as set forth above, NRC regulations allow the use of administrative controls to prevent criticality of fuel in storage. In addition, nothing in the applicable regulations makes a distinction between one time and on-going administrative controls. Further, since human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. To date, there have been no reported incidents of inadvertent criticality in U.S. spent fuel pools for any reason, including violation of administrative procedures.

16. Millstone 3 currently incorporates administrative controls for two-region storage in the existing spent fuel storage racks. These include fuel burnup/enrichment limitations. *See* Millstone 3 Technical Specification (TS) 3.9.13, Figure 3.9-1, Application, Exh. 1, Att. 1 at 14.) TS 3.9.13 and 3.9.14 require surveillances to ensure that all fuel assemblies are properly placed to maintain k_{eff} of the spent fuel pool less than or equal to 0.95 at all times.

Application, Exh. 1, Att. 1). These administrative controls are typical of those discussed above. The administrative controls proposed in the Application, and those contained in supplemental letter of April 17, 2000, only serve to augment the current procedures to the extent necessary to accommodate the 15 new racks and the changes in the regions. Millstone Nuclear Power Station, Unit No. 3, Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98), Apr. 17, 2000 (Exhibit 17). Millstone Unit 3 Surveillance Procedures SP 3866 and SP 31022 are detailed procedures designed to ensure and maintain appropriate boron concentration in the pool, and provide the steps required to ensure spent fuel pool k_{eff} remains less than or equal to 0.95 at all times. Millstone Nuclear Power Station Surveillance Procedure SP 3866, Rev. 3, "Spent Fuel Pool Boron Concentration," Nov. 8, 1996 (Exhibit 18); Millstone Nuclear Power Station Surveillance Procedure SP 31022, Rev. 4, "Spent Fuel Pool Criticality Requirements," Apr. 20, 1997 (Exhibit 19). These administrative controls have not resulted in any reportable instances of fuel assembly misplacements at Millstone Unit 3.

17. Intervenors have stated that there are two classes of administrative measures: those that are made over a finite time and after having been made are no longer necessary (the so called "one time" measures); and administrative measures that are required on an ongoing basis. In reply to the NRC staff's first set of interrogatories, the Intervenors state that "(P)hysical protection against criticality may rely on one-time administrative measures to ensure that the physical protection is in place." CCAM and CAM's Reply to NRC Staff's First Set of Interrogatories, at 9 (April 8, 2000). They further state that, "(F)or example, if physical protection is provided by the geometry of racks, one-time administrative measures will be needed to ensure that the racks are constructed so as to preserve the specified

geometry under specified conditions." *Id.* As stated above, there is nothing in the applicable regulations that supports this theory. Without accepting the Intervenors' theory concerning administrative measures to prevent criticality, we note that the selection of a fuel assembly for storage in Region 1, 2, or 3 of the Millstone 3 spent fuel pool is a one-time administrative measure which, under Intervenors' definition, may be relied upon for physical protection against criticality.

18. The staff has concluded that the above mentioned administrative procedural documentation suffices to ensure that adequate steps are taken to verify that k_{eff} remains less than or equal to 0.95 (5% subcritical) for all regions of the Millstone 3 spent fuel pool. Therefore, the new set of administrative controls does not pose any risk of a criticality accident.

19. Contention 5 states:

> Will the proposed change in schedule of surveillance of the soluble boron in the fuel pool lead to a significantly increased likelihood of a criticality accident stemming from a misloaded fuel element, during the interval between fuel movements?

20. Our response to Contention 5 is stated in the following paragraphs:

21. In a letter to the NRC, dated March 19, 1999, NNECO submitted a request to make changes to selected TS in support of a planned modification of the Millstone 3 spent fuel pool (Exhibit 1). The proposed changes modify the TS to allow for the installation and use of additional storage racks in the Millstone 3 spent fuel pool. Included in this submittal were proposed changes to TS 3.9.1.2 which would have required surveillance of the spent fuel pool boron concentration only during times of fuel movement within the spent fuel pool. Application, Exh. 1, Att. 1, at 1. The minimum required boron concentration would have been changed from its present value of greater than or equal to 1750 parts per million

(ppm), which was required to account for possible Boraflex degradation, to 800 ppm, which is more than sufficient to maintain k_{eff} less than or equal to 0.95 for the worst misloading event. *Id.* Since the Boraflex in the existing storage racks is not being credited in the Licensee's criticality safety analysis, the requirement for 1750 ppm of boron to account for possible Boraflex degradation is no longer necessary. TS 4.9.1.2 would have been changed to require monitoring of the boron concentration every seven days only during fuel movement, rather than every 72 hours when fuel assemblies were in the pool. *Id.*

22. One of the issues raised by the Intervenors was that NNECO's proposal to monitor soluble boron only during fuel movements eliminated a barrier against criticality at Millstone Unit 3.

23. In an April 17, 2000 submission to the NRC, NNECO stated that after evaluation of the February 9, 2000 prehearing conference order (LBP-00-02), it had decided to change the proposed TS amendment to require, in TS 3.9.1.2, that a proposed boron concentration of 800 ppm be maintained at all times when fuel assemblies are stored in the spent fuel pool. Exh. 17 at 2. TS 4.9.1.2 would be amended to require verification of the boron concentration every 7 days. Exh. 17, Att. 1, at 1.

24. NNECO's criticality analysis, performed by Holtec International, for Millstone Unit 3 included several conservative assumptions Holtec International Licensing Report for Spent Fuel Rack Installation at Millstone Nuclear Station Unit 3 (Non-proprietary version), Exh. 1, Att. 5. First, and most notably, the analysis assumed the presence of unborated water. The analysis contained several other conservative assumptions. Racks were assumed to contain the most reactive fuel authorized to be stored in the facility. Unborated water was assumed to be at the temperature yielding the highest reactivity over the

expected range of water temperatures. The analysis assumed an infinite array of storage cells, that is, no neutron leakage (except when assessing peripheral effects and certain accidents). The analysis neglected neutron absorption by minor structural material. In order to maximize the calculated k_{eff}, the analysis incorporated calculational uncertainties and biases, as well uncertainties due to manufacturing tolerances.

25. As part of the NRC review of the NNECO amendment request to establish three regions (Regions 1, 2,and 3) for fuel storage in the spent fuel pool, the Staff reviewed the Holtec report, which presented the criticality evaluation for the misloading of a fresh fuel assembly in the Millstone 3 spent fuel pool. Application, Exh. 1, Att. 5. Based on the analysis described in this report, NNECO has determined that a soluble boron concentration of only 425 ppm would be sufficient to maintain a 5% subcriticality margin in the event of a fuel assembly misloading event (i.e., a fresh PWR assembly enriched to 5 weight-percent U-235 inadvertently loaded into an empty cell in Region 3 with the remainder of the rack fully loaded with fuel of the highest permissible reactivity). The Staff notes that, for conservatism, NNECO has chosen a value of 800 ppm in the proposed TS. Based on experience in evaluating the criticality safety of spent fuel pools, we find the calculational methods and the assumptions made in these analyses to be acceptable and conservative.

26. In addition to the criticality analysis discussed above, NNECO submitted additional Holtec analyses and calculations, discussed below, which demonstrate that criticality is precluded even for various beyond design basis combinations of fresh fuel misplacements and boron dilution.

27. Technical Specification 3.9.1.1 for Millstone 3 requires a minimum boron concentration of 2600 ppm in all filled portions of the reactor coolant system (RCS) and the

refueling canal during refueling operations. Millstone Unit 3 Technical Specification 3.9.1.1. (Exhibit 51). During refueling, the water volume in the spent fuel pool and the refueling canal form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes. Therefore, the actual fuel pool boron concentration is approximately 2600 ppm during refueling operations. As a practical matter, boron in the spent fuel pool does not disappear after fuel movements, nor is it appreciably diluted over time. Any hypothetical event that could dilute the normal boron concentration in the Unit 3 spent fuel pool of approximately 2600 ppm by any significant amount would require such large quantities of water that it would be detectable well before the 800 ppm limit imposed by TS 3.9.1.2 was reached. In addition, high and low water level alarms are present in the fuel handling area and in the control room which would alert the operators to an increasing or decreasing water level in the pool. Increasing or decreasing water levels would also be readily detected visually by plant staff in the spent fuel pool area.

28. Draft Regulatory Guide (RG) 1.13 recommends that the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures. U.S. Nuclear Regulatory Commission, Draft Regulatory Guide 1.13, “Proposed Revision 2 to Regulatory Guide 1.13, ‘Spent Fuel Storage Facility Design-Basis,’” Dec. 1981 (Exhibit 29). This additional safety assurance is based on application of the “double contingency principle” as defined in ANSI/ANS-8.1-1983, which was endorsed by the NRC staff in a generic communication sent to all power reactor licensees on April 14, 1978. ANSI/ANS-8.1-1983, Exh. 16; Letter, B.K. Grimes, NRC, to All Power reactor Licensees, “OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,” Apr. 14, 1978 (Exhibit 27). More recently, the

Commission affirmed this endorsement in 10 C.F.R.§ 72.124(a), which requires at least two unlikely, independent, concurrent or sequential events to have occurred before a nuclear criticality accident is possible. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered one unlikely accident condition and a second concurrent independent accident need not be assumed. Alternatively, credit for the presence of soluble boron in PWR pools may be assumed in evaluating other accident conditions such as the misloading of fresh fuel or fuel that has not attained the required minimum burnup into the proposed Region 1, 2, and 3 storage racks.

29. On June 21, 2000, NNECO supplemented its response to Intervenors' third set of interrogatories by submitting several calculations performed by Holtec International, for the K_{eff} for each of the three Regions in the pool, assuming various misplacements of fresh fuel assemblies in each Region. NNECO's Supplementary Response to CCAM and CAM's Third Set of Interrogatories (June 21, 2000). This analysis includes beyond-design basis criticality calculations involving fresh fuel assembly misplacements and boron dilution. None of these scenarios resulted in criticality in the spent fuel pool. Our review of this document indicates that the calculated criticality values for each Region are correct.

30. The proposed change in the schedule of soluble boron surveillance was rescinded by NNECO's letter of April 17, 2000. Exh. 17. The staff has determined that the results of the Holtec analyses are correct and concludes that a fuel loading error in the proposed burnup-dependent Millstone 3 spent fuel storage racks in Regions 1, 2, and 3 will not lead to a significantly increased likelihood of a criticality accident.

31. Contention 6 states:

> Does GDC 62 permit a licensee to take credit in criticality calculations for enrichment, burn up, and decay time limits, limits that will ultimately be enforced by administrative controls?

32. Our response to Contention 6 follows:

33. NRC regulations (10 CFR Part 50, Appendix A, GDC 62) require that licensees prevent criticality in their spent fuel pools. GDC 62 states 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." A proposed version of the GDC was sent to the Commission in a paper dated June 16, 1967. Memorandum from W.B. McCool to Atomic Energy Commission, "Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits," AEC-R 2/57, June 16, 1967 (Exhibit 22). The AEC (predecessor to the NRC) first formally published the general design criteria for comment on July 11, 1967. Proposed rule, General Design Criteria for Nuclear Power Plant Construction Permits, 32 Fed. Reg. 10,213 (1967)(Exhibit 23). At that time, the proposed criterion for prevention of fuel storage criticality was labeled GDC 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."

34. The AEC received two public comments regarding Criterion 66. The first comment was received from the Oak Ridge National Laboratory (ORNL) on September 6, 1967. Letter, W.B. Cottrell, ORNL, to H.L. Price, Atomic Energy Commission, "Review of USAEC 'General Design Criteria for Nuclear Power Plant Construction Permits,' Federal Register, July 11, 1967," Sept. 6, 1967 (Exhibit 24). Specifically, the ORNL comment on proposed GDC 66 stated that ORNL did not understand the implication of "or processes"

at the end of the first sentence, nor did they believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. They suggested that the last sentence of the criterion should read as follows: "Such means as geometrically safe configurations shall be used to insure that criticality cannot occur." The AEC staff considered these comments and decided that it was not necessary to change the phrase "or processes" and, therefore, it was retained. In considering the second comment, the AEC staff felt that, although the assurance of geometrically safe configurations was the preferable means for preventing criticality, procedural controls should not be ruled out. Therefore, GDC 66 (renumbered GDC 62) was revised to state that geometrically safe configurations are the preferable means for preventing criticality in fuel handling and storage ("Status Report on General Design Criteria," memorandum from Harold L. Price to the Chairman and Commissioners, July 6, 1970, Exhibit 25, "Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969), Exhibit 26). However, it did not specifically rule out other means.

35. The second comment was received from Atomics International. Letter, J.J. Flaherty, Atomics International, to Secretary, U.S. Atomic Energy Commission, Sept. 25, 1967 (Exhibit 30). This comment suggested that the second sentence of the proposed criterion be replaced with "Inherent means should be used where practicable." Although these specific suggested word changes were not incorporated in the final criterion, the AEC did incorporate the intent of the Atomics International comment by stating that geometrically safe configurations (inherent means) are preferred.

36. Fuel reactivity is determined by its physical design, its initial (fresh) enrichment (or weight percent of U-235 to total uranium), and fuel depletion or burnup. The initial

enrichment of a fuel assembly is a physical process consisting of manufacturing an assembly containing a specified weight percent of U-235.

37. Burnup credit is the practice of accounting for the reduced reactivity of spent fuel due to fissile material decay and fission product buildup described above in evaluating criticality safety. The regulations do not elaborate on how or how much subcriticality should be assured, nor do they prohibit the use of burnup credit for criticality safety. As explained above, burnup of fuel occurs as a natural consequence of the fuel's being used in a reactor. Therefore, fuel burnup is a physical process and credit for burnup to prevent criticality in spent fuel storage pools is permitted under GDC 62. Decay time is an extension of the burnup process and includes the time an assembly has been discharged from the reactor and resides in the storage pool. Spent fuel decay time results from the radioactive decay of fissile isotopes in the spent fuel (e.g., U-235) to non-fissile (or non-productive) neutron absorbing isotopes. Thus, the additional decay while cooling in the spent fuel pool further reduces the fuel reactivity. This loss in reactivity due to decay time allows a reduction in the minimum burnup needed to meet the reactivity requirements and typically is applicable to older fuel that has been stored in the spent fuel storage racks for a period of years.

38. The NRC has established a 5% subcriticality margin for wet storage of spent fuel assemblies to assure that licensees meet the requirements of GDC 62. The NRC staff stated this acceptance criterion for criticality in a generic communication sent to all power reactor licensees. Grimes Letter, Exh. 27. This letter states that "The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, <u>including all uncertainties</u>, under all conditions." *Id.*, at III-3. This requirement is also stated in Section 5.6.1.1 of the Millstone 3 TS, which states "The spent fuel storage racks are made up of 3

regions which are designed and shall be maintained ro ensure a K_{eff} less than or equal to 0.95 when flooded with unborated water." Application, Exh. 1, Att. 1, at 25.

39. NNECO proposes to use administrative procedures at Millstone Unit 3 to verify that a fuel assembly has achieved the required amount of burnup to be placed in the proposed storage racks of regions 1, 2, and 3. To date, more than 50 plants have obtained NRC approval for the use of burnup credit for spent fuel storage. The NRC first approved burnup credit in spent fuel pool storage analyses in the early 1980s.[31] Licensees have established their ability to predict core burnup behavior over hundreds of reactor years of operation. They have also established the ability to predict isotopic inventories of reprocessed fuel from data available from several cores of the Yankee reactor. Abstract, R.J. Nodvik, "Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel," WCAP-6068, Westinghouse Electric Corporation, Mar. 1966 (Exhibit 28). Therefore, the NRC has allowed licensees to take credit for burnup in criticality analyses of spent fuel storage pools.

40. The staff considers fuel misplacement in the Millstone 3 regions 1, 2, and 3 storage racks to be an unlikely event for several reasons. First, proposed TS 3.9.13 will control fuel storage limitations and selection procedure. SP 31022, described above, will control fuel assembly selection. Exh. 19. Therefore, both TS as well as plant procedures would have to be violated for a fuel assembly misplacement to occur. In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. Finally,

1. Several plants which were initially approved for burnup credit include Fort Calhoun (1983), St. Lucie 2 (1984), Ginna (1984), Turkey Point 3&4 (1984), and Summer (1984).

the burnup limit curves (Figures 3.9-1, 3.9-3, and 3.9-4) proposed for the Millstone Unit 3 TS for safe storage in regions 1, 2, and 3, are based on minimum burnup requirements. Application, Exh. 1, Att. 1, at 14, 16, 17. These are bounding values that result in just meeting the 5% subcriticality margin for storage pools in region 1, 2, and 3. In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). Such fuel assemblies, therefore, should fall in the acceptable burnup domain specified in these figures, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. Consequently, the placement of a fuel assembly that does not meet the technical specification burnup requirements into a large storage pool, such as the Millstone 3 SFP, and the continued failure to detect this misplacement, is a highly unlikely event at Millstone 3. Multiple misplacements are even more unlikely. It is highly unlikely that a single failure in the administrative controls or the management process will lead to misplacement or multiple misplacements. Such multiple misplacements, with or without boron dilution, leading to criticality, are highly improbable and well beyond the application of the double contingency principle. Although there have been several reported fuel assembly misplacements in spent fuel pools at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in detecting and correcting fuel misloadings. Further, there have been no reported incidents of boron dilution events occurring concurrently with fuel misloading events.

41. In summary, GDC 62 does not prohibit the use of enrichment, burnup or decay time; nor does it prohibit the use of administrative measures to determine if adequate burnup or decay time has been achieved to allow storage in the Millstone 3 spent fuel pool.

NRC STAFF RESPONSES TO BOARD'S QUESTIONS FOR PARTIES

42. On May 23, 2000, the Atomic Safety and Licensing Board (Board) issued a memorandum requesting that the parties address several questions relevant to the three admitted contentions in this proceeding. The NRC Staff hereby responds to the questions posed therein.

Contention 4 (fuel assembly misplacement)

1. Is there any statistical basis for asserting that the likelihood of fuel assembly misplacement is either high or low? If so, what is the basis and what are the statistics? Are this basis and the accompanying statistics applicable to the Millstone-3 reactor or must other factors be taken into account? If there is no such basis, how do you decide on the likelihood of fuel assembly misplacement? (Intervenors, Staff)

STAFF RESPONSE:

No. The Staff is not aware of any statistical study regarding the likelihood of fuel assembly misplacement. Nonetheless, although a quantitative statistical basis has not been established, the number of reported fuel misloading events in spent fuel pools is small in comparison to the large number of fuel movements each year. This indicates that the likelihood of a misloading event is relatively small. There have been reports of fuel misloading events over the years, but many of them are not applicable to spent fuel pool storage since they involved reactor core misloadings or misorientations. Of those applicable to spent fuel storage, most were detected and corrected within a short period of time because of independent checks required by plant procedures, indicating that administrative controls are effective in precluding permanent fuel misloadings. None resulted in any inadvertent criticality events. The Staff has not developed any statistics on the likelihood of a fuel assembly misplacement in a spent fuel pool; however, the Staff considers fuel misplacement in spent fuel storage racks, including those at Millstone Unit 3, to be an unlikely event for the following reasons.

First, proposed TS 3.9.13 will control fuel storage limitations and selection procedure SP 31022, described above (Exh. 19), will control fuel assembly selection. Therefore, both TS as well as plant procedures would have to be violated for a fuel assembly misplacement to occur. In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. Finally, the burnup limit curves (Figures 3.9-1, 3.9-3, and 3.9-4) proposed for the Millstone Unit 3 TS for safe storage in regions 1, 2, and 3, are based on minimum burnup requirements. These are bounding values that result in just meeting the 5% subcriticality margin for storage pools in region 1, 2, and 3.

In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). Such fuel assemblies, therefore, should fall into the acceptable burnup domain specified in these figures, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. Consequently, the placement of a fuel assembly that does not meet the technical specification burnup requirements into a large storage pool, such as the Millstone 3 SFP, and the continued failure to detect this misplacement, is a highly unlikely event at Millstone 3. Multiple misplacements are even more unlikely. It is highly unlikely that a single failure in the administrative controls or the management process will lead to misplacement or multiple misplacements. Such multiple misplacements, with or without boron dilution, leading to criticality, are highly improbable and well beyond the application of the double contingency principle. Although there have been several reported fuel assembly misplacements in spent fuel pools at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in detecting and correcting fuel misloadings.

In the absence of a statistical database to ascertain the consequences of a misplaced assembly, the licensee performs a calculation that analyzes a fuel misloading event involving the misplacement of the most reactive fuel assembly, regardless of the probability of its occurrence. NRC guidance (specifically, the April 14, 1978 letter from B.K. Grimes to all power reactor licensees, at 1.13-9) requires such an analysis to demonstrate compliance with both the Staff's 5 percent subcriticality margin criterion and the subcriticality requirement of General Design Criterion (GDC) 62.

4. When assigning burn-up plus decay time administrative controls, how are the dividing lines between fuel assemblies of various burn-ups decided, and how are uncertainties in burn-up treated? (Licensee, Staff)

<u>STAFF RESPONSE:</u>

The dividing line between fuel assemblies of various burnups is decided based on storage rack composition and configuration. First, the burnup required to allow fuel of the maximum initial enrichment used in the core (typically 5 weight percent U-235) to be safely stored in the racks and meet the 5 percent subcriticality margin is determined. Similar calculations are made for fuel assemblies of lower initial enrichment requiring less burnup to meet the 5 percent margin.

Decay time credit is an extension of this process and includes the time an assembly has been discharged from the core and residing in the pool as a variable. Spent fuel decay time credit results from the radioactive decay of fissile isotopes in the spent fuel to daughter isotopes, which are non-productive neutron absorbers. Thus, the additional decay while cooling in the spent fuel pool further reduces the spent fuel's reactivity. This loss in reactivity due to decay allows a reduction in the minimum burnup needed to meet the reactivity requirements and typically is applicable to older fuel that has been stored in the racks for a period of years.

A large part of the uncertainty in burnup calculations derives from uncertainties in fuel and moderator temperatures and the spectral effect of soluble boron during core operation. Bounding values of these operating parameters are assumed in order to assure the highest plutonium production and, consequently, conservatively high values of reactivity during burnup. The specific code used for burnup calculations is then benchmarked against various critical experiments (including plutonium-bearing fuel), other codes, and reactor operations, to determine calculational bias and uncertainty. In addition, a burnup-dependent uncertainty in reactivity is statistically applied (square root of the sum of the squares) in the determination of the acceptable burnup versus enrichment combinations. Finally, since uniform axial burnup is typically assumed, a correction to account for actual axial burnup distribution is added, if necessary, to the maximum calculated $k_{eff.}$ Thus, all the uncertainties associated with burnup predictions are factored into the calculated k_{eff} to assure that the final k_{eff} is conservative.

<u>CONTENTION 5</u>

> 3. How frequent a check on boron concentration is needed to give adequate assurance of

boron concentration? What determines the needed frequency? (All parties)

STAFF RESPONSE:

The amount of soluble boron in spent fuel pool water is generally determined by the amount of boron required in the reactor core by technical specifications to maintain at least a 5 percent subcriticality margin in the core during refueling. During refueling operations, the water volume in the reactor vessel, the refueling cavity, the refueling canal, and the spent fuel pool form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes. Once the proper amount of soluble boron is installed in the spent fuel pool water, it is very difficult to dilute it. In addition, there are high and low water level alarms in the fuel handling building and in the control room, which, if water were added or lost, would alert the operators to an increasing or decreasing water volume in the pool and, thus, a possible flooding or loss of shielding event. Increasing or decreasing water levels are also readily detectable visually by plant staff in the spent fuel pool area. Any postulated accident condition that could significantly dilute the pool boron concentration would require such large quantities of water (hundreds of thousands of gallons) flowing over the pool and into other areas such as stairwells, that it would be readily detectable long before the soluble boron concentration could be reduced to a level that has any significance with regard to criticality.

Therefore, while there is no specific prescribed frequency for checking soluble boron concentration, based upon the foregoing, the weekly periodic sampling of soluble boron concentration proposed by NNECO is adequate to ensure that any credible dilution accident would be detected and corrected on a timely basis. Finally, although any postulated event that could dilute an appreciable amount of soluble boron is considered to be highly unlikely, a complete loss of all soluble boron was analyzed for the Millstone Unit 3 spent fuel pool and shown to maintain the staff's 5 percent subcriticality margin criterion as well as the GDC 62 requirement for subcriticality.

<u>CONTENTION 6</u> (GDC 62, etc.):

a. Referring to 10 C.F.R. § 50.68(b), what is your definition of reactivity? (Relate this to standard textbook definitions in, <u>e.g.</u>, Hetrick, "Dynamics of Nuclear Reactors," Glasstone & Edlund, "The Elements of Nuclear Reactor Theory," etc.) (All parties)

<u>STAFF RESPONSE:</u>

The definitions used in Hetrick, "Dynamics of Nuclear Reactors," and Glasstone & Edlund, "The Elements of Nuclear Reactor Theory," are the same as that given in most standard nuclear engineering textbooks, including Lamarsh, "Introduction to Nuclear Engineering," 2nd Edition, at 282, that is: (k_{eff} -1) / (k_{eff}). This definition is applied to a system. The definition of this system, in turn, varies depending on the problem of interest. For example, a system might be an individual fuel assembly immersed in pure water or multiple depleted assemblies stored in a poisoned rack with water as moderator. It has been common practice in the nuclear industry to use the phrase "fuel assembly reactivity" to refer to the reactivity worth of a fuel assembly.

2. What is the meaning of the phrase "maximum fuel assembly reactivity" used in 10 C.F.R. § 50.68(b)(4)? How is the maximum fuel assembly reactivity measured? (All parties)

<u>STAFF RESPONSE:</u>

Based on comments received after publication of the proposed 10 C.F.R. § 50.68 in the Federal Register, the Commission replaced the phrase "maximum permissible U-235 enrichment" with "maximum fuel assembly reactivity" since most boiling water reactors (BWRs) have fuel rods of varying enrichments within a bundle and BWR technical specifications are defined in terms of the infinite multiplication factor of an assembly in standard core geometry rather than enrichment. The phrase is also appropriate for pressurized water reactors (PWRs) since enrichment (the percent of U-235 present in the fuel by weight) decreases with fuel burnup. The specification of a maximum enrichment value is only appropriate for fresh fuel. The maximum fuel assembly reactivity is not a measurable quantity but is calculated based on fuel assembly design, initial enrichment, irradiation history in the core, and decay time. It refers to the fuel assembly configuration and nuclide composition that results in the highest k_{eff} when placed in the spent fuel storage racks.

4. When was credit for burn-up first considered in spent fuel pools? Were the considerations involved discussed with the ACRS or the Commission? (Staff)

STAFF RESPONSE:

The NRC first approved burnup credit in spent fuel pool storage criticality safety analyses in the early 1980s. Plants that were initially approved for burnup credit include Fort Calhoun (1983), St. Lucie 2 (1984), Ginna (1984), Turkey Point 3 & 4 (1984), and Summer (1984). It is not known whether the considerations involved were discussed with the Advisory Committee on Reactor Safeguards(ACRS) or the Commission. A memorandum was sent from G. Arlotto to R. Fraley (ACRS) on September 23, 1981, attaching Draft 1 of Regulatory Guide 1.13, Revision 2, for review by the ACRS Regulatory Activities Subcommittee (Exhibit 49). This draft included sections allowing credit for burnup in spent fuel pool storage. However, since the revision was never issued, it is unclear whether it was reviewed by the ACRS.

5. Inasmuch as current spent fuel pool practices appear to have been followed at the time 10 C.F.R. § 50.68 was first formulated, why did the proposed rule or the Statement of Considerations for the final rule not contain an explicit discussion of administrative controls on burn-up and decay time? Why was 10 C.F.R. Part 50, Appendix A, Criterion 62 (GDC 62) (which has been in effect since 1971) not amended or clarified at the time 10 C.F.R. § 50.68 was adopted (during 1998), to reflect that administrative controls on burn-up and decay time (as well as factors explicitly mentioned in GDC 62) could be considered? (Staff)

STAFF RESPONSE:

10 C.F.R. § 50.68 was formulated to give licensees the option of either meeting the criticality accident monitoring requirements of 10 C.F.R. § 70.24 in handling and storage areas for special nuclear material and including criticality accident radiation monitors which would indicate a criticality accident had occurred, or complying with the requirements stated in 10 C.F.R.§ 50.68, which preclude criticality through procedures and design. It was intended primarily for criticality prevention of fresh fuel arrays in handling and temporary storage areas before their storage in fixed geometry racks that are designed to prevent criticality. Therefore, the rule simply states the staff's 5 percent subcriticality criterion for designed spent fuel pool storage racks and does not elaborate on the various acceptable ways to meet the criterion.

The Commission did not need to amend GDC 62 to show that administrative controls could be used since, as written, it does not say that they cannot be used. The

GDCs are general, minimum requirements governing nuclear power plant design and operations. Guidance on how to meet these requirements has been developed over the years by the Staff, primarily through standard review plans (SRPs) and other Staff guidance, such as the Grimes letter (Exhibit 27) and the Memorandum from L.I. Kopp, NRC, to T. Collins, NRC, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," U.S. Nuclear Regulatory Commission (August 1998) (Exhibit 48). Although the staff is not aware of any plans to amend the GDCs, there are plans to update the SRPs. Updates to the fuel storage SRPs will reflect the fact that credit for burnup and decay time and the use of administrative procedures to determine fuel assembly selection and placement are acceptable means for meeting the GDC requirements.

6. What is the scope of the phrase "physical systems or processes" as used in GDC 62? (All parties)

STAFF RESPONSE:

The term "physical systems or processes" refers to the methods available and used to prevent inadvertent criticality in spent fuel pool storage. These systems or processes include geometric separation of assemblies, solid and soluble neutron absorbers, the use of storage racks, and fuel reactivity. Fuel reactivity is determined by fuel assembly design and composition, initial enrichment, and burnup. Fuel burnup, a natural outcome of reactor operation, is also a physical process.

7. Are "procedural controls" included in the scope of "physical processes?" (All parties)

STAFF RESPONSE:

Yes. Implementation of each of the physical processes described above requires some form of administrative controls. For example, administrative controls are used in the fabrication of the storage racks to assure proper geometric spacing and solid neutron absorber placement. Likewise, administrative controls are used when placing fuel assemblies in the storage racks based on accumulated burnup, in order to verify that the required burnup has been attained and that the assemblies are placed in the designated locations.

8. If a licensee changes from an 18-month to a 24-month fuel cycle, what changes must or should the licensee make in the spent-fuel pool? (Intervenors, Staff)

STAFF RESPONSE:

Unless the new, longer operating cycle requires higher enriched fuel than used previously at the plant, no changes are required. If an enrichment increase is made, a new criticality safety analysis is required for the spent fuel storage pool (and fresh fuel storage racks) to confirm that the Staff's 5 percent subcriticality criterion and GDC 62 requirements are met. If the longer cycle results in higher fuel assembly burnup, the discharged fuel will be lower in reactivity than previously discharged fuel and will meet existing Millstone 3 Technical Specification minimum burnup limitations on fuel storage.

43. Exhibits 1, 16-30, 48-51 filed herewith are true and correct copies of the documents relied upon in this affidavit.

44. We both provided the information contained in this affidavit and we hereby certify that the foregoing is true and correct to the best of our knowledge, information and belief.

______________________________ Laurence I. Kopp ______________________________ Anthony C. Attard

Subscribed and sworn to before me
this day of

____
Notary Public

My commission expires:__________________

June 30, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHEAST NUCLEAR ENERGY COMPANY	)	
	)	Docket No. 50-423-LA-3
(Millstone Nuclear Power Station, Unit No. 3)	)	
	)	

AFFIDAVIT OF ANTONE C. CERNE IN SUPPORT OF
THE NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA
AND ARGUMENTS ON WHICH THE STAFF PROPOSES TO RELY
AT ORAL ARGUMENT ON CONTENTION 4

Antone C. Cerne, being duly sworn, does hereby state as follows:

1. I am the Senior Resident Inspector at Millstone Unit 3. I have more than twenty-two years of nuclear experience, including approximately twenty years in the U.S. Nuclear Regulatory Commission's resident inspector program, including the past four years at Millstone Unit 3. Prior assignments were at Seabrook Station and the Pilgrim Nuclear Power Station. At the NRC, I served, on a temporary basis, as Commissioner Kenneth Carr's technical assistant and I have been detailed to the agency's special review effort for Comanche Peak, the NRC Regulatory Review Group, the NRR South Texas Project Task Force, and the NRR group reviewing the Construction Inspection Program for Future Reactors. I have participated in more than thirty NRC team inspections at nuclear power plants around the country, with designation as team leader or assistant team leader on some of these inspections. I was senior resident inspector at Millstone 3 during the conduct of the Independent Corrective Action Verification Program, recovery and startup activities,

and had the responsibility for managing the “significant items list” inspection and closure, as part of the U.S. Nuclear Regulatory Commission’s Inspection Manual Chapter 0350 process. I am qualified as both a Construction and Operations (Pressurized Water Reactor) Senior Resident Inspector. My U.S. Nuclear Regulatory Commission agency-level award recognitions include NRC Resident Inspector of the Year, 1985, the first time the award was given; NRC Meritorious Service Award for Resident Inspector Excellence, 1992; NRC Distinguished Service Award for Senior Resident Inspector Excellence, 1999. I hold a Bachelor of Science degree from the United States Military Academy (West Point),1968, where I was in the top one per cent of my graduating class, and a Master of Science degree in Nuclear Engineering from the Massachusetts Institute of Technology (MIT), 1972. In 1989, I pursued the Program in Science, Technology and Society at MIT on a Mellon (post-graduate) Fellowship. A statement of my professional qualifications is attached hereto as Exhibit 7.

2. The purpose of this affidavit is to address Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone’s Contention 4 as set forth in the Licensing Board’s Memorandum and Order of February 9, 2000. (*Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 3), LBP-00-02, 51 NRC 25 (2000)).

3. The existing spent fuel storage rack design, with its safety evaluation, encompasses two fuel storage regions and considers postulated accidents that could result from the misloading of a fuel assembly in Region 1 or Region 2. Unit 3 Final Safety Analysis Report (FSAR) Section 9.1.2 (Exhibit 8). The “new” administrative controls, as described in the basis for Contention 4, LBP-00-02, 51 NRC at 32, would involve additional controls associated with the addition of a third region of spent fuel racks and the allowance

for two different storage areas in the new Region 1, under the proposed spent fuel rerack design. *See* Millstone Nuclear Power Station, Unit No. 3, Proposed Revision to Technical Specification, Spent Fuel Pool Rerack (TSCR 3-22-98)(Mar. 19, 1999)(Exhibit 1, Att. 3, at 2-4).

4. For both the existing and "new" spent fuel pool rack configurations, the proposed boron concentration in the spent fuel pool would effectively preclude the possibility of a criticality accident caused by a fuel assembly misplacement. *See* Millstone Nuclear Power Station, Unit No. 3 Modification of Proposed Revision to Technical Specification - Spent Fuel Pool Rerack (TSCR 3-22-98) (Apr. 17, 2000)(Exhibit 17).

5. While Contention 4, as admitted, alleges that Northeast Nuclear Energy Company's (NNECO's) application "trades reliance on physical protection for administrative controls," LBP-00-02, 51 NRC at 34, such controls are no greater or more complicated than those required for current operation of the Millstone Unit 3 spent fuel pool. Section 6.8.1 of the Millstone Unit 3 Technical Specifications (TS) states that written procedures shall be established, implemented, and maintained for refueling operations, as well as for typical safety-related activities (included by reference to Appendix A of Regulatory Guide 1.33, Revision 2 (Feb. 1978)) involving operations and other spent fuel pool storage functions. Millstone Unit 3 TS 6.8.1, "Administrative Controls" (Exhibit 9). Section 13.5 of the Millstone Unit 3 FSAR requires plant procedures, including administrative procedures, to control the specifics of station operations. Millstone Unit 3 FSAR Section 13.5 (Exhibit 10). Licensee compliance with these regulatory mandated procedures is a requirement of 10 C.F.R. Part 50, Appendix B, Criterion V, as well as the Unit 3 Technical Specifications.

6. The problems and violations referenced as a basis for Contention 4 are not directly related to Millstone Unit 3. None of the specific violations cited by the NRC in the December 1997 Notice of Violation and Proposed Imposition of Civil Penalties involved spent fuel pool issues at Millstone Unit 3. Notice of Violation and Proposed Imposition of Civil Penalties - $2,100,000 - NRC Inspection Report Nos. 50-245/50-336/50-423: 95-44, 95-82, 96-01, 96-03, 96-04, 96-05, 96-06, 96-08, 96-09, 96-201 (Dec. 10, 1997)(Exhibit 11). Trained personnel can make mistakes, as is discussed by the Licensing Board in admitting Contention 4. LBP-00-02, 51 NRC at 34. This is evidenced at Millstone Unit 3 by two incidents. Plant Information Report No. 3-94-079, "Fuel Misplacement," (Jan. 14, 1991)(Exhibit 12); Adverse Condition Report Transmittal Sheet, ACR # 710, "SFP Crane Operator Went to Wrong Location [;] Stopped by Checker," (Apr. 27, 1995)(Exhibit 13). It is noteworthy, however, that these are the only two personnel errors documented with respect to Unit 3 spent fuel movement issues. Moreover, both errors were identified during the spent fuel movement process and corrected before any assemblies were physically stored in an incorrect location. Report No. 3-94-079, Exh. 12, at 3; ACR #710, Exh. 13, at 3-4. Also, these incidents occurred prior to the 1996 Unit 3 shutdown and initiation of the recovery process that led to a significant Millstone Station culture change and improvements that permitted the authorization of the restart of Unit 3 by the Commission in 1998.

7. Since the restart of Millstone Unit 3 from RFO6 in June 1999, the licensee's performance and overall plant operations have been good, with few operational challenges. NRC inspection results have generally confirmed the safe operation of Unit 3 in accordance with the mandated administrative controls (i.e., approved procedures). NRC Combined Inspection

50-245/99-08, 50-336/99-08, 50-423/99-08 (Sept. 20, 1999)(Exhibit 32); Plant Performance Review - Millstone Unit 3 (Mar. 31, 2000)(Exhibit 40). During any future fuel movements, the licensee is required to implement the applicable, approved procedures to ensure that fuel assemblies with the proper burnup are stored in the correct spent fuel rack locations and that the required double verification process is used to check spent fuel assembly movements to the designated spent fuel racks. 10 C.F.R. Part 50, Appendix B, Criterion V.

8. Reliance upon administrative controls is a given assumption in the safe operation of any nuclear power facility. *See* 10 C.F.R. Part 50, Appendix B; TS 6.8.1 (Exh. 9). The decision at the 2000 NRC Senior Management Meeting to remove Millstone Unit 3 from a regional-focus plant status and return it to normal NRC oversight recognizes that recent licensee performance has demonstrated the ability to properly implement all required administrative controls, including any those needed for spent fuel handling activities associated with the proposed Spent Fuel Pool Rerack at Millstone Unit 3. *See* Transcript, “Commission Briefing on Operating Reactors and Fuel Facilities,” (May 25, 2000), at 9-13 (Exhibit 6).

9. Exhibits 1, 6, 7, 8, 9, 10, 11, 12, 13, 17, 32, and 40 filed herewith are true and correct copies of the documents relied upon in this affidavit.

10. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

____

Antone C. Cerne

Subscribed and sworn to before me
this day of June, 2000.

____

Notary Public

My commission expires: ___________________

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHEAST NUCLEAR ENERGY COMPANY	)	Docket No. 50-423-LA-3
	)	
(Millstone Nuclear Power Station, Unit No. 3)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON CONTENTIONS 4, 5 AND 6" and "AFFIDAVIT OF LAURENCE I. KOPP, AFFIDAVIT OF ANTHONY C. ATTARD, AFFIDAVIT OF JAMES C. LINVILLE, and AFFIDAVIT OF ANTONE C. CERNE" and attached exhibits in the above-captioned proceeding have been served on the following through deposit in the Nuclear Regulatory Commission's internal mail system or, as indicated by an asterisk, hand delivery or as indicated by two asterisks first class mail this 30th day of June 2000.

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_______________________ ***/RA/***
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Counsel for NRC Staff