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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

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

I. INTRODUCTION

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

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SECY-02

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

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

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

II. STATEMENT OF THE CASE

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

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

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

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

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

III. FACTUAL AND PROCEDURAL BACKGROUND

A. History of Criticality Prevention at Nuclear Power Plants

1. Nature of Criticality Accidents

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

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

2. Regulations and agency guidance

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

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

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

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

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

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

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

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

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

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

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

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

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

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

7 *Id.*

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

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

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

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

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

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

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

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

The Kopp Memorandum thus effectively reduces the double contingency principle to a "single contingency principle."
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Thus, as the pressure has increased for higher and higher density fuel storage, the NRC Staff has increasingly relaxed the standards for criticality prevention, allowing the use of administrative controls and reducing the rigor of the accident analysis required.

3. Evolution of Criticality Prevention in Fuel Pools

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

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11 *Id.*, Attachment 4.

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

use of administrative controls for criticality prevention in the high-density storage of spent fuel.

a. Low-density storage

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

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

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

The first step toward higher density was to employ stainless steel racks with center-center distances of about 8 inches in BWR racks and 13 inches in PWR racks. Roughly speaking, this step occurred in the 1970s. The new configuration increased the fuel storage density to a level of up to 0.39 tonne U per square foot. The reduced center-

center distances in this configuration yielded a greater propensity for criticality than was exhibited by the previous open-frame racks. Nevertheless, the rack designers were able to achieve a subcritical margin of reactivity, relying in part on the absorption of slow neutrons by the stainless steel in the rack structures. This neutron-absorption phenomenon was in turn assisted by the moderation of fast neutrons by water confined in passages ("flux traps") between the fuel assemblies. At this stage of evolution in fuel storage density, criticality prevention relied partly on the distance between fuel assemblies and partly on the neutron-absorbing properties of the racks.

c. Incorporation of boron in the structure of storage racks

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

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

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

[14] In pursuit of even higher storage densities in fuel pools, the nuclear industry has also studied fuel storage options involving a reduced presence of water between the fuel rods. Water moderates fast neutrons, so a reduced presence of water can yield a subcritical margin of reactivity even as the spacing between fuel assemblies or rods is reduced. One water-displacing option is to place spent fuel assemblies inside cans and to fill all empty space inside each can with small metal beads, thereby achieving a fuel storage density of 0.75 tonne U per square foot. A second option is to compact fuel assemblies by crushing the fuel

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

d. Ongoing administrative controls

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

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

e. Independent Spent Fuel Storage Installations

There is an alternative to adopting ever-higher densities of fuel storage in an existing fuel pool. That alternative is to construct an independent spent fuel storage

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spacers until rods are nearly touching, thus achieving a fuel storage density of about 0.95 tonne U per square foot. A third option is to dismantle the fuel assemblies and store the rods in close contact with each other inside cans, thus achieving a fuel storage density of about 1.1 tonne U per square foot. None of these options has been generally adopted. *See* U.S. Department of Energy, Spent Fuel Storage Fact Book, DOE/NE-0005 (April 1980).

[15] Burnup is the accumulated fission energy released by a fuel assembly. Its effects on criticality are exploited by restricting the combined burnup/enrichment parameters of fuel assemblies that are placed in

installation ("ISFSI"). ISFSI's have been built at several US nuclear plant sites. In each case, a dry storage technology has been employed. As of September 1998, installations of this kind were licensed at 11 nuclear plant sites.[16]

4. Experience with Administrative Controls

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

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

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

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

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

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the fuel storage racks. Note that in some instances, the reactivity of a fuel assembly will initially increase with burnup, then decrease with higher levels of burnup.

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

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

5. The Millstone Unit 3 Refueling Outage #6

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

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

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

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

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

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

[19] "SM"stands for Shift Manager.

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

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

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

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

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

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

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

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

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

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

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

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

May 16:

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

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

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

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

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

1344 Bypass key used to lower SFP upender.

1410 Bypass key used to lower SFP upender.

1427 Bypass key used to lower SFP upender.

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

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

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

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

The upender problems continued on May 18:

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

1720 Upender will not raise.

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

Problems with the SFP upender continued on May 19:

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

0130 Upender in SFP still unable to raise.

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

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

[23] "RE" refers to reactor engineer.

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

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

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

May 21:

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

June 1:

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

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

May 14:

1155 Loss of communications between CR and all stations.

1215 Communications lost – all Ericksons system went down

May 16: 1225 Ericson communications lost ~ 1 minute.

June 3:

Lost communications, apparently due to Erickson phone network problem.

2150 Gave up on communication – stopped fuel movement.

June 4:

0100 Still no communications.[24]

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

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

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

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

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

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[24] "CR" refers to control room.
[25] "Shuffleworks" refers to a fuel movement system.
[26] "SIGMA" refers to a fuel movement machine in containment.
[27] With so many SIGMA malfunctions, by June 3, log entries concluded that SIGMA had " lost its brain (again)."
[28] The Executive Summary is attached hereto as Exhibit 7.

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

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

The CR further states:

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

The Condition Report is an alarming document.

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[29] The Condition Report, CR-M3-2236 ("Adverse Trend in Performance of the Refueling Equipment"), appears annexed hereto as Exhibit 8.

[30] Citing costs, NNECO may have delayed replacing lightbulbs in the spent fuel pool for as many as four or five years. See Deposition of Michael C. Jensen, Transcript, Exhibit 7, pages 82-83.

[31] Concerns about the expense of maintaining equipment has given rise to concern by some that under deregulation, economic pressures will lead to lessened preventive maintenance being done at nuclear power plants. See Matthew L. Wald, The New York Times, "Con Ed Put Off Plant Upgrade Over Rate Fear," June 30, 2000, page B1, Exhibit 9.

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

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

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

" Apparent Causes

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

[32] A copy of the Beaupre memorandum appears as Exhibit 10.

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

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

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

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

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

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

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

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

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

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

[33] The cover letter appears as Exhibit 11.

B. The Millstone License Amendment Application

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

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

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

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

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

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

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

(CAM), filed a timely request for hearing and intervention petition on October 6, 1999. CCAM is an organization of environmental and safe-energy organizations and individuals based in Mystic, Connecticut. CAM is an organization of environmental and safe-energy organizations and individuals based in East Hampton, New York. On November 17, 1999, in their Supplemental Petition to Intervene, the Intervenors submitted contentions challenging the adequacy of the License Amendment Application.

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

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

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

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

Contention 5 reads as follows:

Contention 5: Significant Increase in Probability of Criticality Accident

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

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

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

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

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

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

And

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

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

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

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

NNECO's application posits that a misloading error may be made in the Millstone Unit 3 spent fuel pool. NNECO's evaluation of such a misloading error determined that a configuration which could yield criticality if it were not for the soluble boron in the water. Yet, NNECO proposes to remove the soluble boron Technical Specification

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

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

without at least providing a surveillance requirement to check for misloaded fuel assemblies at the termination of fuel movements.

Contention 6 reads as follows:

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

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

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

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

GDC 62 requires reliance on physical systems or processes, rather than administrative measures, under both normal conditions and accident conditions. For practical application o GDC 62, a "credible" range of accident conditions must be defined. The NRC has not

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

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

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

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

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

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

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

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

[39]A copy of Interrogatory A2 appears as Exhibit 12.

[40] A set of photographs depicting various views within the spent fuel pool building, including the presence of piping systems, appears annexed hereto as Exhibit 13. They were taken by NNECO during the site inspection on May 10, 2000 at the request of the Intervenors.

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

> A4 Calculations of K-Eff
>
> Given the implementation of the proposed re-racking of the Millstone 3 pool, and assuming an absence of soluble boron, what would be the calculated k-effective in each of the regions of the pool if various combinations of fresh fuel assemblies were placed in the racks? For this purpose, various combinations of fresh fuel assemblies would include one assembly, two adjacent assemblies, four adjacent assemblies, and a full rack, where in each case the surrounding cells would be occupied by assemblies of the highest reactivity allowed by the Technical Specifications.

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

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

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

NNECO's license amendment application proposes the introduction of new administrative controls for the prevention of criticality. This proposed change in the operation of the Millstone Unit 3 fuel pool would significantly increase the probability that a criticality accident or a violation of criticality limits will occur in the pool.[41] The

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

increased probability of a criticality accident would create an undue increase in the burden of risk that arises from operation of the fuel pool. Neither the applicant nor the NRC Staff has a proper understanding of the burden of risk implied by the license amendment, and neither has performed the technical analysis necessary to establish such an understanding. The lack of appropriate analysis provides sufficient grounds for rejection of the license amendment application. Absent such a rejection, the lack of appropriate analysis provides grounds for the conduct of a full evidentiary hearing, in part because the response to CCAM/CAM's discovery was inadequate. Finally, the increased burden of risk would be unnecessary, because alternative modes of spent fuel storage -- especially dry storage, using well-established technology -- are available, allowing the applicant to increase its stock of spent fuel without increasing the risk that arises from operation of the Millstone Unit 3 fuel pool.

A. Significant Increase in the Probability of a Criticality Accident

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

1. Increased Complexity of Administrative Controls

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

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

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

Table IV-1

Allowable Positioning of Fuel in the Millstone Unit 3 Fuel Pool

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

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

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

Notes to Table IV-1:

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

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

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

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

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

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

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

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

and

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

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

3. Criticality Calculations Can Contain Errors

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

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

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

Examples of non-conservative criticality calculations include:

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

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

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

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

[45] A copy of this LER is attached as Exhibit 14.

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

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

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

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

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

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[46] A copy of this LER is attached as Exhibit 15.

[47] A copy of this Supplement is attached as Exhibit 16.

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

4. Fuel Can be Mispositioned

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

Byron Station: May 28, 1996 (Licensee Event Report 454/96-008-00 (June 25, 1996)[48]

On May 28, 1996, three fuel assemblies were found to be present in Region 2 of the spent fuel pool without meeting Technical Specifications requirements. The assemblies did not meet the minimum burnup requirements, nor were they checkerboarded. The required (actual) burnups (in MW-days per tonne U) were: 32,651 (32,648); 32,651 (32,638); and 32,771 (32,728). Two of the three non-complying assemblies were placed in Region 2 in August 1993, and the third assembly was placed in Region 2 in January 1995.

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

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

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

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

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

[48] A copy of this LER is attached as Exhibit 17.
[49] A copy of this LER is attached as Exhibit 18.

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

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

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

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

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

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

[50] A copy of this LER is attached as Exhibit 19.
[51] A copy of this LER is attached as Exhibit 20.

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

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

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

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

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

[52] So NNECO's Michael C. Jensen acknowledged at his deposition. Exhibit 8, page 19.

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

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

5. Dilution of Soluble Boron Can Occur

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

[53] The incident is one of the reported events documented in Exhibit 21; it was discussed during the Jensen deposition. See Exhibit 7, pages 60-66.
[54] Exhibit 7, page 63.
[55] A copy of pages 48-48 of the Reactor Engineering log RFO 5 is annexed as Exhibit 22.

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

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

B. Undue Increase in the Burden of Risk

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

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

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

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

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[56] Robert Griffin, Millstone Chemistry Manager, during his May 12, 2000, deposition, described an incident at Millstone Unit 2 when hydraulic fluid entered the spent fuel pool. The liquid contents of the pool were put through an osmotic process for purification. Griffin Deposition Transcript, Exhibit 23, page 128.

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

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

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

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

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

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

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

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

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

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

A I have not yet.

Q What else do you need to look at?

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

Q Haven't looked at what?

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

Q You have never looked at some of these?

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

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[59] Anthony C. Attard Deposition Transcript, Exhibit 24, page 5.
[60] Exhibit 24, page 10.
[61] Exhibit 24, pages 11-12.

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

A No, not after this, no.

Q What about the Grimes letter?

A No. None o these –

Q The particular ANSI standards that you referred to?

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

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

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

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

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

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

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

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[62] Exhibit 24, page 14
[63] Exhibit 24, page 15
[64] Exhibit 24, pages 17-18

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

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

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

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

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

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

[65] Exhibit 24, pages 20-21
[66] Laurence I Kopp Deposition Transcript, Exhibit 25, page 24
[67] Exhibit 25, page 26
[68] Exhibit 25, pages 12-16, 21

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

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

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

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

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

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

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[69] Exhibit 25, page 36
[70] Exhibit 25, page 40
[71] Exhibit 25, page 43
[72] Exhibit 25, page 43

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

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

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

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

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

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

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

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[73] Exhibit 25, pages 45-46
[74] See Robert Griffin Deposition Transcript, Exhibit 16, pages 63-66.
[75] Exhibit 25, pages 47-49
[76] Exhibit 25, page 55
[77] Exhibit 25, page 56
[78] Exhibit 25, pages 64-66

3. Technical Analysis is Insufficient

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

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

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

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

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

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[79] See NNECO's Supplementary Response to Intervenors' Third Set of Interrogatories dated June 21, 2000, Exhibit 26.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Similarly, if restrictions on fuel burnup/enrichment or fuel age are to be exploited as means of criticality suppression in a rack in a fuel pool, then ongoing administrative controls must ensure that a fuel assembly is never placed in the rack unless its burnup/enrichment or age is within a specified range. Ongoing administrative controls on fuel burnup/enrichment or fuel age can be specified for an entire pool, for a particular rack, or for particular spaces within a rack. At a number of nuclear plants, a

"checkerboard" pattern of fuel placement has been specified, wherein particular spaces in the repeating checkerboard pattern have particular restrictions on fuel burnup/enrichment. These administrative controls must be effective on each occasion when a fuel assembly could be placed in the pool.

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

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

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

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

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82 A copy of this Information Notice is attached as Exhibit 27.

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

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

1. Pre-rulemaking documents

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

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

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

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

84 The Press Release and attached documents are attached as Exhibit 28.

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

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

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

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

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

2. Proposed GDC for criticality control

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

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85 Internal AEC memorandum from G.A. Arlotto to J.J. DiNuuno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966), attached as Exhibit 29.

86 The October 25, 1966, letter and attached draft are attached to this Summary as Exhibit 30.

87 The February 8, 1967 letter and attached draft are attached to this Summary as Exhibit 31.

88 Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967). The Note and relevant excerpts from Appendix B to the Note are attached as Exhibit 32.

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

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

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

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

3. Comments on the proposed rule

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

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

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89 A copy of the Federal Register notice is attached to this Summary as Exhibit 33.

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

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

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

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

4. The Final Rule

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[90] ORNL's comments on the proposed rule were contained in an attachment to a letter of September 6, 1967 from William B. Cottrell of ORNL to H. L. Price of the AEC, attached as Exhibit 34.

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

[92] Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Units (July 15, 1969), attached as Exhibit 35.

[93] *See* Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970) Attached to the Wiggin letter is a marked-up version of the June 4, 1966, revised draft of the GDC. The Case Memorandum and enclosed documents are attached as Exhibit 36.

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

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

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

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

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

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[94] Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971). A copy of the Federal Register notice is attached as Exhibit 37.

1. 10 C.F.R. §§ 70.24 and 50.68

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

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

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

[95] Proposed Rule, Criticality Accident Requirements, 62 Fed. Reg. 63,911; Direct Final Rule With Opportunity to Comment, Criticality Accident Requirements, 62 Fed. Reg. 63,825.
[96] 62 Fed. Reg. at 63,825, Col. 3.

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

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

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

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

The language of § 50.68, as it was finally promulgated, contains a list of measures for criticality prevention that can be implemented in lieu of maintaining a criticality

monitoring system.[98] Although these provisions contain some references to procedures and administrative measures, they do not undermine or contradict the general requirement of GDC 62 for physical criticality prevention measures. For instance, subsection (b)(1) requires that:

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

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

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

> (2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage rack shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.
>
> (3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2. 10 C.F.R. § 72.124

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

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

[102] Exhibit 26.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Astoundingly, the Staff has pursued this course for over two decades without conducting any safety analysis to determine whether its radical departure from the requirements of GDC 62 could be justified on safety grounds. The Staff has never done a systematic analysis of the potential for criticality accidents when reliance is placed on administrative measures instead of physical measures. Although the Staff has advocated the Double Contingency Principle in evaluating criticality accidents since 1978, it has

made no attempt to determine what combinations of fuel handling or pool management errors would violate the Double Contingency Principle. Instead, as discussed above and in Appendix A, it has merely watered down the Double Contingency Principle to a Single Contingency Principle. Despite the many years of accumulated licensee experience with spent and fresh fuel storage, the Staff has never attempted to conduct a systematic review of the operating experience of licensees with fuel mispositioning or fuel incidents relevant to boron dilution.[108] The Staff does not even maintain a systematic data base of the experience of nuclear power plant licensees with such problems as mispositioning of fuel assemblies and soluble boron management errors.

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

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

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

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

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

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

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

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

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relevant events. *See* Appendix C.

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

VII. CONCLUSION

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

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

Respectfully submitted,

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Counsel to Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone

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

(Millstone Nuclear Power Station, Unit No. 3) ASLBP No. 00-771-01-LA

DECLARATION OF DR. GORDON THOMPSON IN SUPPORT OF INTERVENORS' SUMMARY AND SWORN SUBMISSION REGARDING CONTENTIONS 4, 5 AND 6

I, Gordon Thompson, declare as follows:

1. I am the executive director of the Institute for Resource and Security Studies (IRSS), a nonprofit, tax-exempt corporation based in Massachusetts. Our office is located at 27 Ellsworth Avenue, Cambridge, MA 02139. IRSS was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace and international security, efficient use of natural resources, and protection of the environment.

2. I received an undergraduate education in science and mechanical engineering at the University of New South Wales, in Australia. Subsequently, I pursued graduate studies at Oxford University and received from that institution a Doctorate of Philosophy in mathematics in 1973, for analyses of plasma undergoing thermonuclear fusion. During my graduate studies I was associated with the fusion research program of the UK Atomic Energy Authority.

3. During my professional career, I have performed technical and policy analyses on a range of issues related to international security, energy supply, environmental protection, and sustainable use of natural resources. Since 1977, a significant part of my work has consisted of technical analyses of safety and environmental issues related to nuclear facilities. These analyses have been sponsored by a variety of nongovernmental organizations and local, state and national governments, predominantly in North America and Western Europe. Drawing upon these analyses, I have provided expert testimony in legal and regulatory proceedings, and have served on committees advising US governmental agencies. A copy of my resume is appended as Exhibit 4 to the CCAM/CAM Supplemental Petition to Intervene (November 17, 1999).

4. I have reviewed the March 19, 1999 license amendment application filed by the Northeast Nuclear Energy Company (NNECO) for an amendment to Facility

Operating License No. NFP-49, which seeks permission to expand the storage capacity of the Millstone Unit No. 3 spent fuel pool. I have also reviewed pertinent portions of the Final Safety Analysis Report for the Millstone Nuclear Power Station, and various correspondence and technical documents relating to the proposed amendment and to risks of spent fuel storage, which are identified in the Intervenors' contentions. In addition, I attended a site inspection of the Millstone Unit 3 spent fuel pool on May 10, 2000.

5. I participated in the preparation of the Intervenors' contentions regarding the proposed license amendment. Following admission of Contentions 4, 5 and 6, I assisted in evaluating whether NNECO's license amendment application poses an undue and unnecessary risk of a criticality accident, increases the probability of a criticality accident and whether it conforms to the requirements of General Design Criterion 62 and applicable NRC Staff guidance.

6. In making my evaluation, I conducted an extensive review of documents related to criticality prevention at Millstone and in general, including correspondence between NNECO and the NRC Staff, criticality studies performed by or for NNECO, NRC Staff and licensee documents regarding proposed spent fuel storage expansion applications, Licensee Event Reports of criticality-related occurrences, NRC Staff and industry guidance documents and related correspondence, the rulemaking history of GDC 62, and other publicly available information regarding spent fuel storage and criticality prevention. I also participated in preparing for depositions of NNECO and NRC Staff witnesses regarding Contentions 4, 5 and 6, and in reviewing the deposition testimony of these witnesses. In addition, I was deposed by NNECO.

7. I have contributed to the development of the technical factual assertions contained in Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone Intend to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Millstone Unit No. 3 Nuclear Power Plant, including Appendices A, B and C, submitted to the Licensing Board on June 30, 2000 (hereinafter "Summary"). My contributions to the Summary have been true and correct to the best of my knowledge and professional judgment.

I declare, under penalty of perjury, that the foregoing is true and correct.

Executed on June 30, 2000.

G.R. Thompson

Gordon Thompson

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

I hereby certify that copies of "Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which Connecticut Coalition Against Millstone and Long Island Coalition Against Millstone Intend to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact With the Licensee Regarding the Proposed Expansion of the Spent Fuel Storage Capacity at the Millstone Unit 3 Nuclear Power Plant, including Appendices A, B and C" in the above-captioned proceeding have been served on the following by E-Mail as indicated by asterisk and to all by conforming copy via U.S. Mail, postage pre-paid, on July 3, 2000:

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Appendix A

The Double Contingency Principle

1. Introduction

In addressing the potential for inadvertent criticality in spent fuel pools, the Nuclear Regulatory Commission (NRC) Staff and the American Nuclear Society (ANS) have employed the concept of a "double contingency principle". This appendix describes and compares the versions of this concept that have been articulated by the NRC Staff and the ANS.

2. The Grimes Letter

In 1978, the NRC Staff issued guidance for spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications." The guidance was attached as Enclosure No. 1 to an April 14, 1978 letter from Brian K Grimes to "All Power Reactor Licensees." This letter and its enclosures are hereafter described as the "Grimes letter". In addressing the potential for a criticality accident, the Grimes letter states:

> "The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident."

Id., Enclosure 1 at page III-1.

Thus, the Grimes letter states that a criticality analysis must demonstrate that two unlikely, independent, concurrent events must occur before there is a criticality accident.

Immediately following the statement quoted above, the Grimes letter goes on to suggest that:

> "Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies."

The concept of "realistic initial conditions" is not defined in the Grimes letter, and is therefore open to interpretation. It is not plausible that the authors of the Grimes letter intended to say that soluble boron concentrations will never fall below their specified level. Instead, the Grimes Letter reasonably presumes that,

at the outset of an accident sequence, conditions in the spent fuel pool will be in a "normal" range.

Any sequence of events that leads to a criticality accident in a fuel pool will have an end point, namely the criticality event. By projecting backward in time from the end point, one will always be able to identify an earlier point in time at which the pool's characteristics were in their normal range. For example, at this earlier point, the concentration of soluble boron in the pool water would have been as specified by licensee procedures or Tech Specs. One could reasonably describe the conditions at the earlier point in time as realistic initial conditions.

As a sequence of events unfolds toward a criticality accident, conditions will change in a manner specific to that sequence. For example, the concentration of soluble boron in the pool water might fall, and this occurrence might be preceded or followed by placement in the pool of fuel assemblies with a higher-than-specified reactivity. Alternatively, an earthquake or the falling of a large object into the pool might reduce the center-center distance in the fuel racks. To apply the double contingency principle, as articulated in the Grimes letter, one must identify "events" of this kind and determine if they are "unlikely", "independent" and "concurrent".

3. Draft Regulatory Guide 1.13

The double contingency principle was re-stated and revised in Appendix A of Proposed Revision 2 to the NRC staff's Draft Regulatory Guide 1.13, dated December 1981, titled "Spent Fuel Storage Facility Design Basis". Paragraph 1.4 of Appendix A 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 <u>not</u> occur without at least two unlikely, independent, and concurrent failures or operating limit violations."

This paragraph is broadly consistent with the statement of the double contingency principle in the Grimes letter, but there are two notable differences. First, Paragraph 1.4 specifies "at least two" criticality-inducing events, whereas the Grimes letter specifies "two" events. This difference significantly strengthens the double contingency principle, as explained below. Second, Paragraph 1.4 refers to "failures or operating limit violations" whereas the Grimes letter refers to "events".

The Draft Reg. Guide's use of the phrase "at least two" to modify the number of failures or violations that must be considered is significant, because it indicates that the drafters of the guidance were concerned about identifying potential interactions of causative events (failures or violations), beyond a single occurrence.[1] Thus, if a combination of two causative events is shown to cause criticality, and there is any possible doubt about the events being unlikely, independent and concurrent, then the Draft Reg. Guide indicates that this occurrence of criticality would be unacceptable.

Similarly, by referring to "failures or operating limit violations" rather than "events", the Draft Reg. Guide makes the double contingency principle more useful, by giving clearer guidance regarding the events that must be considered.

4. A Definition by the American Nuclear Society

The ANS has provided a definition of the double contingency principle, although not specifically in the context of fuel management. This definition appears in ANS Standard ANSI/ANS-8.1-1983, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors", approved October 7, 1983 and reaffirmed November 30, 1988. It should be noted that ANSI/ANS-8.1-1983 was endorsed by Revision 2 to the NRC staff's Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities", dated March 1986.

ANSI/ANS-8.1-1983 defines the double contingency principle as follows:

> "Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible."

Id. at page 3.

Note that ANSI/ANS-8.1-1983 is a revision of ANSI N16.1-1975, which is the ANSI standard that is cited in the Grimes letter.

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[1] Appendix C describes how a fuel pool's envelope of criticality can be determined. This envelope bounds the combinations of events that can cause criticality. Determining the envelope of criticality is a necessary precursor to applying the double contingency principle.

5. Another Statement by the American Nuclear Society

A statement of the double contingency principle appears in ANS Standard ANSI/ANS-57.2-1983, "American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants", approved October 7, 1983. In addressing the scope of criticality safety assessment, ANSI/ANS-57.2-1983 states:

> "At all locations where spent fuel is handled or stored, the nuclear criticality safety analysis shall demonstrate the criticality could not occur without at least two unlikely, independent and concurrent incidents or abnormal occurrences."

Id., Paragraph 6.4.2.1.4.

Similar language appears in ANS Standard ANSI/ANS-8.17-1984, "American National Standard Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors", approved January 13, 1984, reaffirmed March 20, 1997. ANSI/ANS-8.17-1984 states:

> "The fuel unit and rods should be handled, stored and transported in a manner providing a sufficient factor of safety to require at least two unlikely, independent, and concurrent changes in conditions before a criticality accident is possible."

Id., Paragraph 4.11.

In addressing the role of neutron-absorbing materials, such as boron, in preventing criticality, ANSI/ANS-8.17-1984 states:

> "Reliance may be placed on neutron-absorbing materials, such as gadolinium and boron, that are incorporated in the fuel material itself, or in structures or equipment, or in both. However, when reliance is placed on neutron-absorbing materials, control shall be exercised to maintain their continued presence with the intended distributions and concentrations. Extraordinary care should be taken with solutions of absorbers because of the difficulty of exercising such control and with fuel units containing burnable poison to identify the maximum reactivity condition to be considered."

Id., Paragraph 4.9.

ANSI/ANS-57.2-1983 provides specific guidance regarding the assumptions about soluble boron that should be made in a criticality analysis. At Paragraph 6.4.2.2.9, ANSI/ANS-57.2-1983 states:

> "The presence of a soluble neutron absorber in the pool water shall not be considered in the evaluation of ks for PC I, II and III. In the analysis for PC IV and V faults, the <u>initial</u> presence of soluble neutron absorber may be assumed, if it is normally used, until addition of unborated makeup begins."

(emphasis in original)

In this context, ks is the evaluated maximum neutron multiplication factor in the fuel racks. Plant Conditions (PC) I through V are defined at pages 2-3 of ANSI/ANS-57.2-1983. PC I events are "those events that are expected to occur regularly or frequently in the course of normal operation at the facility". PC II events are those with an estimated frequency of a least 1 per 10 reactor-years. PC III events are those with an estimated frequency of at least 1 per 100 reactor-years but less than 1 per 10 reactor-years. An example of a PC III event would be a loss of offsite power for up to 8 hours. PC IV and V events "are not expected to occur during the life of the facility, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material". Their estimated frequency is between 1 per 1 million reactor-years and 1 per 100 reactor-years. An example of a PC IV or V event would be a loss of offsite power for up to 7 days.

6. A Current Interpretation by the NRC Staff

In recent years the NRC staff has articulated, and used for licensing purposes, a particular interpretation of the double contingency principle. This interpretation is set forth in a regulatory guidance document attached to an internal NRC Staff memorandum by Laurence Kopp to Timothy Collins, dated August 19, 1998 (hereafter known as the "Kopp memorandum"). The Kopp memorandum articulates the double contingency principle as follows:

> "The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be

> assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions."

Kopp memorandum at page 4.

This interpretation has been employed by the NRC staff in approving amendments to operating licenses for a number of nuclear power plants. In illustration, consider the NRC's issuance on June 29, 1998 of Amendments No. 102 and No. 80, respectively, to the operating licenses for Vogtle Units 1 and 2 (Facility Operating Licenses NPF-68 and NPF-81). Those amendments allowed an increase in Vogtle Unit 1 spent fuel storage capacity from 288 to 1,476 assemblies. The NRC Staff's accompanying Safety Evaluation Report addressed criticality analysis in the context of potential accidents, and indicated that the double contingency principle can be applied in that context. The report states:

> "However, for such events, the double contingency principle can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident."

Id. at page 5.

The Kopp memorandum's articulation of the double contingency principle differs significantly from the statement in the Draft Reg. Guide, because it does not require the consideration of "at least two" unlikely, independent and concurrent events." It also substitutes the word "events" for the Draft Reg. Guide's instruction to consider "failures or operating limit violations," thereby returning to the less-useful language of the Grimes letter.

Moreover, the Kopp memorandum provides incorrect guidance regarding the need to consider reductions in the concentration of soluble boron in the pool water. In the excerpt quoted above, the Kopp memorandum states that "credit for the presence of the soluble boron may be assumed in evaluating other accident conditions". This statement is incorrect because there could be situations in which a reduced concentration of soluble boron, occurring in combination with one other failure (e.g., the mispositioning of some fuel assemblies), causes criticality without the other failure being unlikely,

independent and concurrent. The other failure might be likely (i.e., the "unlikely" requirement is not satisfied), might share an underlying cause with the reduced concentration of soluble boron (i.e., the "independent" requirement is not satisfied), or might precede or follow the reduction in soluble boron concentration (i.e., the "concurrent" requirement is not satisfied). In any of those situations, the Kopp memorandum would provide incorrect guidance.

7. A Comparison of the Various NRC and ANS Interpretations

The sources cited here show two schools of interpretation of the double contingency principle. The first school of interpretation encompasses the Grimes letter, Draft Regulatory Guide 1.13, and the relevant ANS standards. The second school of interpretation encompasses the Kopp memorandum and the current licensing practice of the NRC Staff.

The first school says that at least two abnormal events must occur before there is criticality.[2] The second school says that a criticality accident is acceptable if it follows just one abnormal event. Moreover, the Kopp memorandum incorrectly advises that the presence of soluble boron can always be assumed in evaluating the potential for another event to lead to criticality.

Overall, the second school provides a significantly weaker standard of protection against inadvertent criticality. This divergence between the two schools is much more significant than the comparatively minor divergences of interpretation that exist within the first school.

Within the first school, the most detailed guidance for application of the double contingency principle is provided by ANSI/ANS-57.2-1983. This document provides, as described above in Section 5, specific guidance about the assumptions that should be made regarding the presence of soluble boron.

The guidance in ANSI/ANS-57.2-1983 may be useful, insofar as it does not conflict with the full application of the double contingency principle, as set forth in effectively identical language in Draft Reg. Guide 1.13 and ANSI/ANS-57.2-1983. Full application of the double contingency principle requires the determination of the envelope of criticality for the fuel pool in question, and the

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[2] The Grimes letter takes a minority position within the first school by not requiring "at least two" abnormal events. This discrepancy could be ascribed to the relatively early date of the Grimes letter. At that time, the complexities of criticality analysis may not have been fully appreciated.

systematic evaluation of events represented in that envelope to determine if they are unlikely, independent and concurrent.

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Appendix B

Some Incidents Relevant to the Potential for Criticality in Fuel Pools

INTRODUCTION

This appendix describes a variety of incidents at US nuclear power plants. Each incident sheds light on the potential for inadvertent criticality in spent fuel pools. A feature common to all the incidents is that they involve a failure of administrative controls.

The original sources of information on the incidents described here are Licensee Event Reports (LERs) and related documents. This source material was obtained through discovery in license proceedings and through other channels.

The historical record summarized here is almost certainly incomplete, for three reasons. First, the source documents used here were not selected through a search of the full, US-wide body of LERs and related documents, and therefore address a subset of the relevant, documented incidents.[1] Second, each relevant incident that has been identified by a nuclear plant licensee was not necessarily documented by submission of an LER or otherwise. Third, it is highly likely that a significant number of relevant incidents have occurred but have not been identified by the responsible licensee.

The remainder of this appendix consists of a set of incident descriptions. The incident descriptions are in five categories. Within each category, the descriptions are arranged by alphabetic order of the plants where the incidents occurred. The incident categories are:

(1) Incidents involving fuel mispositioning;
(2) Incidents involving spent fuel pool operations;
(3) Incidents involving criticality calculations;
(4) Incidents involving management of soluble boron; and
(5) Miscellaneous incidents.

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[1] A comprehensive search of relevant, available records would have required resources beyond those available to CCAM, and has not been performed by any other entity.

(1) INCIDENTS INVOLVING FUEL MISPOSITIONING

Braidwood Unit 1: July 10, 1996 (Licensee Event Report 456/96-008-00 (August 5, 1996))[2]

During the verification of spent fuel pool storage locations, it was discovered on July 10, 1996 that one fuel assembly stored in Region 2 did not comply with a Tech Spec requirement that the assembly should be stored in a checkerboard configuration, based on its burnup level. Contrary to that requirement, the assembly was stored in a close-packed configuration.

The non-complying fuel assembly had been discharged from the reactor core on October 11, 1991 and relocated to Region 2 of the pool on June 16, 1992. Initially, its storage configuration met Tech Spec requirements for burnup. Those requirements became more stringent on January 20, 1995, at which time the assembly should have been relocated to Region 1 or to a checkerboard configuration in Region 2. Neither step was taken, because the burnup of this assembly was incorrectly entered into a spreadsheet program that was used to determine if assemblies were stored appropriately. The spreadsheet calculations were not independently verified.

Braidwood Unit 1: June 17, 1996 (Licensee Event Report 456/96-007-00 (July 15, 1996))[3]

On June 17, 1996, while spent fuel assemblies were being repositioned in the spent fuel pool, the Fuel Handling Supervisor noted a fuel configuration in Region 2 of the pool that had a potential for criticality that was not bounded by the existing criticality analysis. This configuration had been specified by the Nuclear Material Custodian on May 9, 1996, and the configuration had then been accepted by two independent reviewers, on May 11, 1996 and May 15, 1996. The licensee attributed this incident to personnel error, and to procedural and management deficiencies.

Neither the number of assemblies involved in this incident, nor the details of the configuration, are stated in LER 456/96-007-00. The potentially critical configuration involved the interface between: (a) fuel whose burnup level allowed it to be placed at any location in Region 2; and (b) fuel whose burnup

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[2] A copy of this LER is attached as Exhibit B-1.

[3] A copy of this LER is attached as Exhibit B-2.

level required that it be checkerboarded. Calculations performed for the licensee indicated that criticality in this configuration would be suppressed by the presence of soluble boron in the pool water at a concentration exceeding 300 ppm.

In addition, the LER reports that a licensee review of plant records revealed one previous instance of fuel mispositioning. In that instance, fresh fuel was mispositioned in the spent fuel pool during transfer from the New Fuel Storage Vault. The cause was attributed to "personnel error due to a lack of a questioning attitude and failure to follow procedures."

Browns Ferry Unit 2: September 14, 1980 (Licensee Event Report (October 9, 1980))[4]

During a refuelling outage, two fuel assemblies in the core were found to be rotated 90 degrees from their correct orientation. These two assemblies were among sixteen assemblies that had been loaded with an incorrect orientation during the previous refuelling outage. During that outage the incorrect orientation was detected for each of the sixteen assemblies, but was corrected for only fourteen assemblies. Thus, two assemblies remained in an incorrect orientation until the next outage.

Byron Station: May 28, 1996 (Licensee Event Report 454/96-008-00 (June 25, 1996))[5]

On May 28, 1996, three fuel assemblies were found to be present in Region 2 of the spent fuel pool without meeting Tech Spec requirements. The assemblies did not meet the minimum burnup requirements, nor were they checkerboarded. The required (actual) burnups (in MW-days per tonne U) were: 32,651 (32,648); 32,651 (32,638); and 32,771 (32,728). Two of the three non-complying assemblies were placed in Region 2 in August 1993, and the third assembly was placed in Region 2 in January 1995.

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

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4 A copy of this LER is attached as Exhibit B-3.

5 A copy of this LER is attached as Exhibit B-4.

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

Byron Station: July 15, 1994 (Licensee Event Report 454/94-006-00 (August 15, 1994))[6]

On July 15, 1994, one fuel assembly was found to be present in Region 2 of the spent fuel pool without meeting Tech Spec requirements. The assembly did not meet the minimum burnup requirements, nor was it checkerboarded. The required (actual) burnup (in MW-days per tonne U) was: 32,540 (29,770). The non-complying assembly was placed in Region 2 in September 1993.

The Nuclear Materials Custodian (NMC) mistakenly allocated two non-complying fuel assemblies for placement in Region 2. This mistake arose because inappropriate procedures were used for assembly allocation. A reviewing engineer detected the NMC's mistake for one fuel assembly but not the other. The reviewing engineer's failure to detect both of the NMC's mistakes arose from the reviewing engineer's use of inappropriate procedures.

Cooper Station: November 18, 1986 (Licensee Event Report 298/86-034-00 (December 18, 1986))[7]

On November 18, 1986, during a refuelling outage, it was discovered that fresh fuel with a U-235 loading in excess of the Tech Spec limit had been stored in the spent fuel pool during three cycles of plant operation. The Tech Spec limit on U-235 loading was 14.5 grams per axial centimeter.

During Cycle 7, fresh fuel with a U-235 loading slightly higher than the Tech Spec limit was stored in the spent fuel pool between February 3, 1981 and April 27, 1981. The same phenomenon occurred during Cycle 10, between July 23, 1984 and July 17, 1985. During Cycle 11, fresh fuel with a U-235 loading of 14.6 grams per axial centimeter was stored in the spent fuel pool for some period

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6 A copy of this LER is attached as Exhibit B-5.

7 A copy of this LER is attached as Exhibit B-6.

prior to the determination on November 18, 1986 that the Tech Spec limit had been violated.

The Tech Spec limit of 14.5 grams per axial centimeter on U-235 loading was introduced in June 1978 as part of Tech Spec amendments that provided for installation of high-density fuel racks in the spent fuel pool. Criticality calculations performed at that time were based on a fuel design for which the U-235 loading was 14.5 grams per axial centimeter.

Crystal River Unit 3: November 9, 1987 (Licensee Event Report 302/87-026-00 (December 1, 1987))[8]

On November 9, 1987, the reactor vessel was completely defuelled. It was discovered that a fresh fuel assembly with a U-235 enrichment of 3.85% had been placed in the "A" spent fuel pool. The Tech Spec limit on the enrichment of fuel in the "A" pool was 3.5%.

This event occurred because a mistaken entry was made on a Fuel/Control Assembly Move Sheet. The intention was to move an assembly from location M42 in the "B" spent fuel pool to the "A" spent fuel pool. The assembly in location M42 would have complied with the Tech Spec requirements for placement in the "A" pool. Location M43 was mistakenly entered on the Move Sheet, leading to transfer of the non-complying fresh fuel assembly from the "B" pool to the "A" pool. This transfer was detected about 80 minutes after its occurrence.

Davis-Besse Station: October 20, 1994 (INPO OE 7220, Fuel Mispositioning During Refueling (April 11, 1995))[9]

During refuelling on October 20, 1994, a fuel assembly was unintentionally moved from one core position to another. Step 518 of the Fuel Movement Sequence Sheet called for the fuel assembly in position R-08 to be removed. The convention was "R" stood for "Reactor" as opposed to "S" for "Spent Fuel Pool" and "O8" meant core position O-8. The main fuel handling bridge operator went to core position R-8 instead. The spotter reported to the fuel handling director that the bridge was at "Revelations" meaning R-8. The fuel handling director, functioning as the independent verifier, did not notice the error. The fuel assembly in position R-8 was moved to core position F-10.

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[8] A copy of this LER is attached as Exhibit B-7.

[9] A copy of this document is attached as Exhibit B-8.

The error was detected later when another fuel assembly was positioned over core location O-8. The bridge operator noticed that location O-8 was not empty.

The root cause evaluation of this event concluded the error was due to "the failure of the independent verification by the fuel handling director." The evaluation identified a contributing factor to be "overconfidence due to a history of success."

Farley Unit 1: March 23, 2000 (Licensee Event Report 348/2000-004-00 (April 20, 2000))[10]

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

The plant's licensee reported: "Manual verification of the acceptability of the proposed offload configuration on March 11, 2000 failed to identify that three assemblies had insufficient burnup for their planned storage locations." The licensee reported that the error occurred because: "As the SFP [spent fuel pool] approaches capacity with time, the complexity of the task of determining acceptable storage configuration has increased."

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

Hope Creek Station: December 12, 1995 (Licensee Event Report 354/95-042-00 (March 25, 1996))[11]

On December 12, 1995, during a refuelling outage, a visual inspection of the reactor core revealed that one fuel assembly was 180 degrees out of its proper orientation. The mis-oriented assembly had not been moved since its emplacement on April 3, 1994. A visual inspection of the core had been

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10 A copy of this LER is attached as Exhibit B-9.

11 A copy of this LER is attached as Exhibit B-10.

performed at the time of emplacement, using a video camera. This inspection had not detected the mis-orientation of the assembly. A previous mis-orientation at Hope Creek had been detected during post-emplacement inspection.

McGuire Unit 1: October 24, 1991 (Licensee Event Report 369/91-016-00 (November 25, 1991))[12]

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

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

Oconee Unit 1: January 8, 1996 (Licensee Event Report 269/96-001-00 (February 7, 1996))[13]

On December 14, 1995, a fuel assembly was lifted from its location in the spent fuel pool, so that the assembly could be visually inspected. After the inspection, the assembly remained suspended from the refuelling bridge. This situation was discovered on January 8, 1996 by two fuel handlers who were starting preparations for loading a dry cask some days later.

The two fuel handlers proceeded to lower the suspended assembly into the open location immediately beneath the assembly. Their intention was to allow an identification of the assembly in order to determine its correct location and to trace its previous movements. Through this action the fuel handlers returned the assembly to its location of December 14, 1995, although they did not know this prior to lowering the assembly.

The licensee reviewed previous operating experience, industry-wide and at the Oconee site, in an effort to identify related incidents. Findings from this review

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12 A copy of this LER is attached as Exhibit B-11.

13 A copy of this LER is attached as Exhibit B-12.

were summarized in Attachment A of LER 269/96-001-00, but with limited supporting detail. Some of the information in Attachment A is excerpted in the following two paragraphs.

Four related NRC Level IV Violations were recorded at Oconee in the period 1992-1995, as follows: (a) in November 1990, a fuel assembly was placed in a wrong location in the reactor core; (b) a similar event occurred in February 1993; (c) in September 1991, a fuel assembly was placed in an incorrect location in the spent fuel pool; and (d) in August 1994, a refuelling sequence was altered without proper documentation and procedural control, and a fuel assembly was retrieved from an incorrect location in the spent fuel pool and placed in the reactor core.

Related incidents identified from industry-wide experience included: (a) several fresh fuel assemblies were received and placed in incorrect rack locations; (b) six fuel assembly mispositioning events occurred during refuelling and defuelling operations; (c) unauthorized movement of a defective, encapsulated spent fuel rod occurred; (d) four events occurred which involved inadequate oversight of refuelling operations and inadequate performance by refuelling personnel; (e) a control rod was inserted in the wrong fuel assembly; and (f) six events occurred that involved human performance deficiencies while reactor core components were being handled.

Oyster Creek Unit 1: January 21, 1987 (Licensee Event Report 219/87-006-00 (February 24, 1987))[14]

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

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

[14] A copy of this LER is attached as Exhibit B-13.

fully reloaded (on September 14, 1986), all of the fresh fuel had been removed from the spent fuel pool.

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

Susquehanna Unit 1: October 6, 1993 (NRC Information Notice 94-13 (February 22, 1994))[15]

During reactor defuelling operations, personnel performing the fuel handling activities removed an incorrect fuel assembly from a peripheral location in the reactor core. On becoming aware of this error, the personnel involved returned the assembly to its prior position in the core. That action was contrary to licensee procedures, which required that: (a) the assembly was to be placed in the spent fuel pool; and (b) fuel handling activities were to be halted until the cause of the error was determined and corrected.

Waterford Station: February 18, 1994 (NRC Information Notice 94-13, Supplement 1 (June 28, 1994))[16]

While the reactor was at 100% power, an unknown object was found hanging from the fuel-handling machine in the fuel-handling building. The object was subsequently identified as a capsule containing a defective fuel rod that had been removed from an irradiated fuel assembly several years earlier and then stored in a rack in the spent fuel pool.

Licensee investigations suggested that the capsule had become attached to the fuel-handling machine during unauthorized use of the machine between February 11 and February 18, 1994. The licensee speculated that one of the people assigned to prepare for a March 1994 refuelling outage had inadvertently lifted the capsule while practicing the use of the hoist. No keys or special knowledge were needed to operate the fuel-handling machine. None of the personnel questioned by the licensee admitted to unauthorized use of the machine.

This Information Notice offered some suggestions to licensees to prevent unauthorized or unintended use of fuel-handling equipment, including locking

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15 A copy of this Information Notice is attached as Exhibit B-14.

16 A copy of this Supplement is attached as Exhibit B-15.

circuit breakers in a deenergized position and placing placards that warn against unauthorized use.

(2) INCIDENTS INVOLVING SPENT FUEL POOL OPERATIONS

Arkansas Nuclear One Unit 2: March 20, 1996 (NREC Morning Report 4-96-0028 (March 21, 1996))[17]

On March 20, 1996, workers at ANO Unit 2 aligned the spent fuel pool purification system from the refuelling water tank to the spent fuel pool in order to replace a filter in the purification system. Shortly afterwards, a low spent fuel pool level alarm was received in the control room. Operators responding to the alarm discovered a valve that was supposed to be fully closed, but was partially open. Approximately 900 gallons of water drained from the spent fuel pool to the radwaste boron management system tank. The level in the spent fuel pool dropped 1.5 inches.

Calvert Cliffs Unit 2: April 3, 1997 (NRC Information Notice 97-68, Loss of Control of Diver in a Spent Fuel Storage Pool (September 3, 1997))[18]

On April 3, 1997, refueling at Calvert Cliffs Unit 2 was suspended due to a malfunction of the fuel transfer system. Three dives had been made into the reactor cavity to repair the equipment. A fourth dive was made, this time into the south end of the spent fuel pool to work on equipment from that side. All of the dives were controlled by approved procedures and a job-specific radiation work permit. The dives into the reactor cavity had been monitored via underwater cameras. The dive into the spent fuel pool was monitored by a spotter on the refuelling floor.

As the diver waited for replacement parts to be obtained and lowered into the pool to repair a limit switch, he radioed to the support team for permission to inspect a kink in the transfer cable. The Radiation Protection technician approved the request, mistakenly assuming that the kink was in the cable at the south end of the spent fuel pool. The diver swam to the north end of the spent fuel pool. The spotter was distracted from observing the diver and did not see the diver leave the authorized work area. The diver swam to within a few feet of radiation fields of 12,000 to 20,000 rad/hour.

Upon the diver exiting the pool, the support team realized that the diver had left the work area authorized by the radiation work permit and may have been over-

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17 A copy of this document is attached as Exhibit B-16.

18 A copy of this Information Notice is attached as Exhibit B-17.

exposed. Despite recognition that the administrative controls intended to prevent over-exposure had failed, the diver was sent into the spent fuel pool for a fifth dive before his dosimetry was checked to determine the extent of his exposure.

The NRC's Information Notice documented numerous opportunities to prevent the over-exposure, to limit the over-exposure once the problem occurred, and to prevent recurrence. All of these opportunities were missed.

Cooper Station; McGuire Units 1 and 2; Millstone 1; North Anna Units 1 and 2; Oconee Units 1, 2 and 3; South Texas Units 1 and 2; Summer; Turkey Point Units 3 and 4; and Vogtle Unit 1: (NRC Press Release 96-74, NRC Staff Completes Survey of Refueling Practices at Nation's Nuclear Power Plants (May 22, 1996))[19]

The NRC staff conducted a nationwide survey of refueling practices. Table 2 of the staff's report (dated May 21, 1996) to the NRC Commissioners listed 15 nuclear units that may have conducted past refuelling outages in violation of their licensing bases.

Thus, the actual practices during fuel movements at these fifteen nuclear units did not ensure compliance with the requirements in the Tech Specs and updated Final Safety Analysis reports.

Diablo Canyon Unit 1: February 25, 1999 (NRC Information Notice 99-21, Recent Plant Events Caused By Human Performance Errors (June 25, 1999))[20]

Around 1 PM on February 25, 1999, workers replaced relay CIAX-H (which is associated with the containment Phase A isolation signal) per an approved work order. Approximately four hours later, the operators in the main control room received a spent fuel pool alarm. An operator responding to the alarm discovered that the spent fuel pool cooling pump was not operating. Investigation determined that the relay replacement tripped the spent fuel pool cooling pump, consistent with a design objective to prevent overloading the emergency diesel generators during an accident. The clearance issued for the work order did not recognize that the relay replacement would trip the spent fuel pool cooling pumps and the operators were not aware that spent fuel pool cooling had been lost until the alarm sounded. The spent fuel pool heatup rate

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19 A copy of this document is attached as Exhibit B-18.

20 A copy of this Information Notice is attached as Exhibit B-19.

was calculated to be 8 degrees F per hour, with boiling conditions reached in about 16 hours.

Haddam Neck: August 14, 1985 (Licensee Event Report 50-213/87-015-00 (September 11, 1987))[21]

An electrician troubleshooting binding problems with a relay on August 14, 1985, pushed the manual trip latch mechanism. That action resulted in the complete loss of two electrical buses. Operators took immediate actions to restore power to the buses, but smoke from the electrical panel housing the undervoltage relays for the buses caused operators to manually trip the bus breakers.

Haddam Neck was in refuelling at the time. The loss of the two buses caused spent fuel pool cooling to be lost. In addition, an irradiated fuel assembly was suspended from the fuel handling crane, which was powered from the de-energized buses.

Actions to restore power to one of the buses took 80 minutes. The spent fuel pool water temperature increased 6 degrees F during that time.

This event revealed another administrative control weakness: "Due to problems experienced on the initial classification of this event, no prompt report was made pursuant to the requirements of 10CFR50.72." The licensee did not report this event within 14 or 30 days as required by 10 CFR. Instead, the licensee waited until after the NRC reminded it of the reporting requirements.

Hatch Unit 1: December 28, 1994 (INPO SER 15-95, Spent Fuel Pool Liner Punctured by Dropped Equipment (June 25, 1995))[22]

During the process of removing seven defective core shroud head bolts from the spent fuel pool on December 28, 1994, the stainless steel wire rope sling used to lift the first bolt failed. The bolt, which had been raised out of the water by about a foot, fell back into the spent fuel pool, glanced off the side of the pool, and punctured the 3/16-inch thick floor liner plate. The bolt was 3 inches in diameter and 17 feet long and weighed 365 pounds.

Workers moved to the second bolt without realizing that the first bolt had punctured the spent fuel pool liner. The spent fuel pool water level dropped

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21 A copy of this LER is attached as Exhibit B-20.

22 A copy of this document is attached as Exhibit B-21.

about 2 inches in 23 minutes, triggering an alarm in the control room. An operator responding to the alarm saw the lowered spent fuel pool level and the first bolt protruding from the floor. Approximately 2,000 gallons of water drained from the spent fuel pool to the radwaste system before the operator closed a valve in the drain line. The bolt was removed from the pool on December 31, 1994, and a temporary rubber patch placed over the hole until a permanent patch could be welded in place three days later.

Hope Creek Station: (NRC Enforcement Action 98-308 (June 4, 1998))[23]

On June 4, 1998, the NRC issued a Notice of Violation (NOV) to the owner of the Hope Creek plant for several violations. The first violation cited in the NOV involved the refuelling outage conducted at Hope Creek in 1990. The reactor core had been fully off-loaded to the spent fuel pool. The plant's design and licensing bases rely on the residual heat removal (RHR) system to augment spent fuel pool cooling capability during full-core offload conditions. However, the RHR system was removed from service. The NRC determined that an Unreviewed Safety Question (USQ) was introduced during the 1990 refueling outage, but the plant's owner had neither sought nor obtained NRC approval.

Susquehanna Unit 2: August 22, 1995 (INPO OE 7443, New Fuel Bundle Dropped in Spent Fuel Pool (August 25, 1995))[24]

As new fuel assemblies were being transferred from the Unit 2 New Fuel Vault to the Fuel Prep Machine in the spent fuel pool on August 22, 1995, a fuel assembly fell 15 to 20 feet into the Fuel Prep Machine. The fuel assembly went through the top guide of the Fuel Prep Machine and impacted the lower carriage support plate. The drop occurred when the general purpose grapple detached from the jib crane cable swivel connector.

Millstone Unit 1: March 1, 1996 (NRC Daily Event Report 30050 (March 1, 1996))[25]

During a pre-job briefing on moving the spent fuel pool gates out of the Millstone 1 spent fuel pool, it was determined that the load path would violated Procedure MP 790-4, "Control of Heavy Loads." Specifically, the gate would be

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23 A copy of this document is attached as Exhibit B-22.
24 A copy of this document is attached as Exhibit B-23.
25 A copy of this document is attached as Exhibit B-24.

transported over irradiated fuel assemblies and an analysis of the consequences from a gate dropping onto the fuel had not been performed.

Investigation determined that the procedure had been developed in the early 1980s. The spent fuel pool was reracked in 1988, putting the racks close enough to the spent fuel pool walls so that compliance with NUREG-0612 (Control of Heavy Loads) might not be possible.

This event demonstrated that effective administrative controls can be rendered ineffective if they are not updated to reflect current plant configurations.

Millstone Unit 1: September 17, 1993 (Licensee Event Report 245/93-011-02 (July 25, 1996))[26]

The licensee of Millstone Unit 1 reported that "during prior refueling outages the spent fuel pool cooling system, by itself, would have been incapable of maintaining pool temperature below the 150 degree Fahrenheit design limit, under certain conditions. The conditions in question involve the transfer of a full reactor core into the spent fuel pool."

The licensee of Millstone Unit 1 reported that the "certain conditions" were satisfied nine (9) times in the past: September 1974 during refueling outage (RFO) 2, September 1975 during RFO-3, October 1976 during RFO-4, March 1978 during RFO-5, October 1980 during RFO-7, September 1982 during RFO-8, April 1984 during RFO-9, October 1985 during RFO-10, and June 1987 during RFO-11.

This event revealed how administrative controls intended to ensure that safety margins were maintained were violated repeatedly over a 12 year span. It can be reasonably assumed that one individual did not make the same mistake nine times. Instead, there were many mistakes by many people.

Millstone Unit 1: October 18, 1993 (NRC Information Notice 95-54, Decay Heat Management Practices During Refueling Outages (December 1, 1995))[27]

On October 18, 1993, the licensee of Millstone Unit 1 reported that it had determined that conditions may have existed during which the spent fuel pool cooling system may have been incapable of maintaining spent fuel pool temperature below the 150 degrees F design limit for continued operation.

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26 A copy of this LER is attached as Exhibit B-25.

27 A copy of this document is attached as Exhibit B-26.

Specifically, the plant owner reported that: (1) inappropriate assumptions had been made in the analysis performed for a spent fuel pool re-rack project in 1988; (2) the "normal" refueling sequence described in the updated Final Safety Analysis Report assumed offload of only one third of a core; (3) the FSAR considered a full-core offload as an "emergency" (or abnormal discharge) offload whereas full-core offloads were normally performed at Millstone Unit 1; and (4) under certain circumstances Millstone Unit 1 may have operated outside its design basis for the spent fuel pool.

Thus, as a result of inappropriate assumptions in calculations and failure to comply with administrative controls, conditions in the Millstone Unit 1 spent fuel pool may have been outside their design bases.

Millstone Unit 3: June 26, 1997 (NRC PNO-I-97-039, Spent Fuel Pool Heatup (June 27, 1997))[28]

While recording plant conditions on June 26, 1997, an operator noticed that the Millstone Unit 3 spent fuel pool temperature was higher than it had been at the last reading. Further investigation revealed that the reactor plant component cooling water (CCW) system was lined up to the wrong spent fuel pool heat exchanger. Water from the Unit 3 spent fuel pool was flowing through the heat exchanger that was not being supplied cooling water by the CCW system. The system misalignment had occurred 28 hours earlier. The spent fuel pool water temperature increased from 87 degrees F to 98 degrees F during the period when no cooling was provided.

This event showed the good and bad aspects of administrative controls. Spent fuel pool conditions were monitored at least once every eight hours. The problem was detected during one of these periodic checks. However, the problem was missed during at least two prior checks.

Susquehanna Unit 1: September 12, 1987 (INPO OE 2473 (March 16, 1988))[29]

Prior to a refueling outage, workers removed gates to cross-connect the Unit 1 and Unit 2 spent fuel pools at Susquehanna. Prior to flooding the reactor cavity of Unit 1 during a refueling outage, workers performed a leak test on the reactor cavity seals. After the test was completed on September 12, 1987, workers returned valves to positions in accordance with the test procedure. Because the

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[28] A copy of this document is attached as Exhibit B-27.

[29] A copy of this document is attached as Exhibit B-28.

test procedure assumed that the fuel pool gates would be installed, the drain valves on the cask storage pit were opened. Approximately 11,000 gallons of water drained from the cross-connected spent fuel pools until alarms were received from the fuel pool cooling system.

This event demonstrated that administrative controls are only effective when applied to appropriate configurations. Workers complied with the administrative controls of the test procedure, but nevertheless created an unintended leakage pathway for spent fuel pool water.

Various plants and incidents: (NRC Information Notice 94-13 (February 22, 1994))[30]

Various fuel-handling incidents occurred at Vermont Yankee, Peach Bottom, Susquehanna and Nine Mile Point Stations during the period September-November 1993. This Information Notice drew a generic lesson as follows:

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

Various spent fuel pools: (AEOD Presentation to NRC Commissioners, Assessment of Spent Fuel Cooling (November 14, 1996))[31]

Following reports of design problems at Susquehanna and Millstone, the NRC's AEOD staff examined reported problems at US nuclear power plants over an approximate 12 year period (1984-1996). The AEOD reported 38 actual events in which water was lost from a spent fuel pool and 55 other "near-misses." The AEOD reported 56 actual events in which cooling of a spent fuel pool was lost and 22 other "near-misses."

The AEOD concluded that the probability of losing spent fuel pool cooling was 2 to 3 per 1,000 reactor years and the probability of losing a foot or more of spent fuel pool water level was 1 per 100 reactor years.

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30 A copy of this Information Notice is attached as Exhibit B-29.

31 A copy of this document is attached as Exhibit B-30.

The AEOD concluded that the likelihood of problems was "highly dependent on human performance and plant design."

Vermont Yankee: September 9, 1993 (NRC Information Notice 94-13 (February 22, 1994))[32]

On September 9, 1993, the NRC sent an Augmented Inspection Team (AIT) to Vermont Yankee after a series of events involving irradiated fuel assemblies. On September 3, 1993, an operator had lowered the fuel grapple onto an irradiated fuel assembly in the reactor core. As he lifted the fuel assembly out of the core, it slipped off the fuel grapple and fell about eight feet back into the reactor core. About ten minutes later, the radiation readings for the air flow leaving the main stack nearly doubled. The operators evacuated personnel from the drywell as a precaution and declared an Unusual Event, the least severe of the four emergency conditions.

The AIT concluded that the "fingers" of the fuel grapple had not properly closed on the fuel assembly baling handle. The operator had not rotated the fuel grapple after latching the fuel assembly as directed by procedure. When a fuel assembly is properly latched, the fuel grapple cannot freely rotate. The improperly latched fuel assembly slid off the lower "fingers" of the fuel grapple and plunged back into the reactor core.

On September 9, 1993, an operator lowered a fuel assembly, latched onto the fuel grapple, until it struck another fuel assembly in the reactor core. The operator wanted to raise the fuel assembly, but had accidentally depressed the LOWER pushbutton instead.

The AIT concluded that mistakes made by refuelling personnel were the immediate causes of both events. In addition, weaknesses in the human factors aspects of the controls for the fuel handling equipment contributed to the event in which a fuel assembly was lowered rather than raised. The controls for the fuel handling equipment had been modified shortly before this event occurred. The team concluded that the root cause of the events was a significant weakness in management oversight of fuel handling activities. Weak management oversight had allowed many of the measures intended to prevent a fuel handling accident to be neglected."

Thus, human error and management weaknesses caused the administrative controls intended to prevent a fuel handling accident to be neglected at this nuclear plant.

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32 A copy of this Information Notice is attached as Exhibit B-31.

Yankee Nuclear Power Station: March 15, 2000 (Licensee Event Report 029/2000-02 (April 13, 2000))[33]

During preparations to conduct spent fuel inspections in the spent fuel pool., it was discovered on March 15, 2000, that the past practice used in moving spent fuel from the lower tier racks to the upper tier racks was contrary to the design basis as described in the Final Safety Analysis Report (FSAR). The FSAR states that the storage racks are designed to maintain proper spacing and structural integrity after being impacted by a fuel assembly dropped onto any location from a height of six inches above the top of the racks. Contrary to that design basis, past practice allowed spent fuel movement over the lower tier racks to a height of approximately 13 inches above the top of the racks.

The 6-inch restriction became part of the plant's design bases as a result of Technical Specification Proposed Change #131 dated September 25, 1975. The NRC's Safety Evaluation Report approving the amendment request was dated December 29, 1976. Subsequently, Yankee Atomic Electric Company (YAEC) increased the fuel storage capacity in the early 1980s by adding an upper tier of racks. YAEC reported: "The practice of moving spent fuel assemblies in accordance with OP-7107 required an implied movement of the assembly higher than 6 inches to move it up and over the lower tier grating. While a precaution in the procedure clearly stated the 6-inch height restriction, the procedure was silent on the steps and pathway necessary to move the assembly up the additional 7 inches and over the grating and preserve the 6-inch travel height restriction. Since it was literally impossible to move the assembly over a partially grated fuel rack without lifting the assembly higher than the 6-inch limitation, the practice was to move the assembly higher in a path directly over vacant fuel storage cells in order to move it up and over the grating."

YAEC had a design bases requirement not to lift fuel assemblies higher than 6 inches above the top of the storage racks and a precaution in their controlling procedure about it, yet fuel movements were conducted for more than a decade contrary to the design bases and procedural requirements.

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33 A copy of this LER is attached as Exhibit B-32.

(3) INCIDENTS INVOLVING CRITICALITY CALCULATIONS

Braidwood Unit 1: August 21, 1996 and March 25, 1997 (Licensee Event Report 456/96-010-02 (August 11, 1998))[34]

On August 21, 1996, an analysis of blackness test[35] data was received by the licensee, indicating shrinkage and gaps in the Boraflex in the spent fuel racks. The largest gap exceeded the dimensions that had been assumed in the then-current criticality analysis. This situation arose because of deterioration of the Boraflex. In response, the licensee initiated the process of requesting a license amendment to allow credit for soluble boron as a means of criticality control.

On March 25, 1997, a modelling deficiency was identified in a criticality analysis dated October 31, 1996. That analysis had incorrectly assumed that Boral poison panels are located on all four faces of all storage cells in Region 1 of the spent fuel pool. The same assumption had been carried forward through successive criticality analyses since 1987. In fact, the peripheral Region 1 cells do not have Boral panels on their exterior faces.

McGuire Units 1 and 2: March 2, 2000 (Licensee Event Report 369/00-03 (March 30, 2000))[36]

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

Criticality calculations for McGuire had been performed using 2-dimensional geometry. It had been believed that the 2-dimensional calculations were conservative enough to easily bound fuel assemblies stored in the spent fuel pools. However, the 3-dimensional calculations using non-uniform fuel assembly axial burnup distributions showed that the results from the 2-dimensional calculations were non-conservative for lower burnups and

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34 A copy of this LER is attached as Exhibit B-33.

35 Blackness testing is a technique in which a neutron source is used to evaluate the degradation of Boraflex neutron-absorbing material in spent fuel storage racks.

36 A copy of this LER is attached as Exhibit B-34.

enrichments. The plant's licensee concluded that: "Given the actual fuel assembly burnups and the existing limits, the potential existed that k_{eff} would exceed 0.95 under the postulated unborated condition."

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

Millstone Unit 2: February 14, 1992 (Licensee Event Report 336/92-003-01 (June 25, 1992))[37]

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

Millstone Unit 2: (NRC Information Notice 92-21, Supplement 1, Spent Fuel Pool Reactivity Calculations (April 22, 1992))[38]

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

Thus, this report demonstrated that spent fuel pool criticality margins can be adversely impacted by calculation errors.

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37 A copy of this LER is attached as Exhibit B-35.

38 A copy of this Supplement is attached as Exhibit B-36.

(4) INCIDENTS INVOLVING MANAGEMENT OF SOLUBLE BORON

Catawba Unit 1: March 5, 1990 (Licensee Event Report 413/90-016-00 (April 19, 1990))[39]

The Boric Acid Tank (BAT) and the Refueling Water Storage Tank (FWST) were major sources of borated water at the plant. On February 5, 1990 the plant's Chemistry Department was informed by operations personnel that the BAT was the declared source of borated water. From February 5 through February 26, 1990, the Chemistry Department took samples from the BAT and the FWST, to comply with Tech Spec requirements.

During the period March 5 through March 12, 1990, the Chemistry Department failed to take a sample from the FWST as required by the Tech Specs. During that period the Chemistry Department continued to believe that the BAT was the declared source of borated water. On March 14, 1990 the Chemistry Department contacted operations personnel to confirm this belief, but was informed that the BAT had been inoperable since March 1, 1990.

The licensee attributed this incident to personnel error and deficient communication between departments.

McGuire Unit 1: July 11, 1994 (Licensee Event Report 369/94-005-00 (August 10, 1994))[40]

On July 10, 1994, while the reactor was at 100% power, plant personnel began to drain the spent fuel pool transfer canal. During the drain-down, a water misting system was used to keep the walls of the transfer canal wet to minimize potential airborne contamination. This misting system added demineralized, un-borated water to the transfer canal. During the drain-down, the spent fuel pool was separated from the transfer canal by a gate. Drain-down was accomplished by lowering a submersible pump into the transfer canal. It appears that the discharge from the submersible pump was directed into the pool.

By a route not specified in LER 369/94-005-00 (but presumably via the submersible pump), approximately 28,000 gallons of demineralized, un-borated water were added to the spent fuel pool during the drain-down process. This

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39 A copy of this LER is attached as Exhibit B-37.

40 A copy of this LER is attached as Exhibit B-38.

occurred on July 10 and 11, 1994. According to measurements performed on July 12, 1994, the addition of the demineralized water to the pool had lowered the soluble boron concentration in the pool from 2,105 ppm to 1,957 ppm. The Tech Specs require a boron concentration in the pool of 2,000 ppm.

The licensee attributed this incident to a variety of personnel errors and procedural deficiencies. The LER states: "Personnel interviewed did not have a good understanding of their responsibilities associated with Reactivity Management."

Three Mile Island Unit 1: February 4, 1998 (Licensee Event Report 289/98-002-01 (April 3, 1998))[41]

Tech Specs at this plant require sampling of spent fuel pool water for soluble boron content, both monthly and between 24 to 48 hours after completion of each water addition. On January 23, 1998, water was added to the pool between 0918 and 1705 hours, but no sample was subsequently taken within the specified time period. A further water addition was made on January 27, 1998 between 1410 and 1817 hours. The pool was then sampled at 0430 hours on 28 January 1998 and again at 0830 hours on January 29, 1998. On February 4, 1998 a Staff Chemist noticed that this sampling sequence did not meet Tech Spec requirements for timely sampling after the January 23 water addition.
The licensee attributed this incident to personnel error and the absence of a warning sign that was supposed to be attached to the wall directly behind the valve used to fill the spent fuel pool. The missing sign would have reminded personnel to notify the Chemistry Department of the need for sampling.

A previous failure to perform sampling after a water addition to the pool had occurred in June 1996. In response to that failure, the licensee had modified the plant procedures. One of the modifications was to require placement of a warning sign -- the same sign that was absent in January 1998.

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41 A copy of this LER is attached as Exhibit B-39.

(5) MISCELLANEOUS INCIDENTS

Beaver Valley Units 1 and 2: (NRC Enforcement Action EA 97-076 (March 24, 1997))[42]

On March 24, 1997, the NRC issued a Notice of Violation (NOV) to the licensee of Beaver Valley Units 1 and 2 for three violations involving: (1) numerous failures to follow procedures and implement appropriate work practices and controls, resulting in numerous valves and switches being in the wrong position; (2) operators inadvertently deenergizing the waste gas decay tank (WGDT) oxygen analyzers because of operator error, in that the operators misoperated the oxygen analyzer control switches and also failed to monitor oxygen concentration when running the degassifier to the WGDT; and (3) failure to take appropriate corrective action, despite numerous configuration control deficiencies being identified in 1995, to ensure that the general problem of configuration control was adequately corrected.

The NRC did not impose a civil penalty with this NOV, because the agency had "recently issued a $160,000 civil penalty to [Beaver Valley] on March 10, 1997, for, in part, a violation that involved a valve being in the wrong position for several years despite prior opportunities to identify and correct this condition."

Thus administrative controls were repetitively violated over an extended period of time despite several opportunities to correct the non-compliance problems.

Braidwood Unit 2: 1991 (NRC Report AEOD/T94-02, Review of Mispositioned Equipment Events (March 17, 1994))[43]

An operator attempted to replace a seal filter on Unit 2. The filter was partially ejected from its housing and radioactive water spilled into the room. The problem was caused by failure to valve out the filter prior to the replacement effort. The individual sent to independently verify that the filter was valved out mistakenly went to the Unit 1 room. The Unit 1 filter was not valved out. The independent verifier could not observe the valve position because he had not brought along proper anti-contamination clothing and attempted to perform the verification from behind the radiological barrier.

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42 A copy of this document is attached as Exhibit B-40.

43 A copy of this document is attached as Exhibit B-41.

This event demonstrated that independent verification, or double-checking, does not always prevent errors. In this case, the independent verifier made two errors -- going to the wrong unit, and not seeing that the valve was open instead of closed.

According to this NRC report, there were over 190 mispositioning event reported by US plant owners from 1990 to 1993. From its review of nearly two hundred events, the NRC concluded: "The independent verification process mean to catch mispositioned equipment is not always successful."

Cooper Station: (NRC Enforcement Action 96-202 (September 30, 1996))[44]

On September 30, 1996, the NRC issued a Notice of Violation (NOV) to the licensee of the Cooper Nuclear Station for failures by licensed operating personnel to follow procedural requirements, including: (1) a failure to insert control rods in the proper sequence following a loss of operation of a reactor recirculation pump; (2) a failure to notify shift supervision of an unexpected situation, i.e., a mispositioned control rod, for approximately 20 minutes; and (3) a failure to obtain the concurrence of the shift supervisor and reactor engineer in developing a recovery plan for a mispositioned control rod.

Thus, licensed operators at this nuclear power plant violated administrative controls on reactivity manipulations for the reactor core and compounded that violation by not seeking proper advice and authorization to correct the non-conforming condition.

Zion Unit 1: February 21, 1997 (NRC Enforcement Actions 97-222 and 97-223 (September 2, 1997))[45]

On September 2, 1997, the NRC imposed a civil penalty of $330,000 on the licensee of the Zion nuclear plant. One-third of that civil penalty was assessed for a reactivity management event on February 21, 1997. The plant was being shut down, pursuant to Tech Specs, due to an inoperable containment spray pump. Management failed to effectively communicate to the operating staff their expectations for shutting down the reactor. Management assumed that Unit 1 was being shut down since the containment spray pump could not be repaired within the Tech Spec-allowed outage time. However, the shift engineer directed

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44 A copy of this document is attached as Exhibit B-42.

45 A copy of this document is attached as Exhibit B-43.

that the unit be kept critical in anticipation of the pump being returned to service.

The licensed reactor operator assigned to reduce reactor power to the point of adding heat inadvertently made the reactor subcritical. When the operator realized that the reactor was substantially subcritical -- instead of stopping, evaluating, and communicating the unauthorized change in reactivity -- he started withdrawing rods to make the reactor critical at the point of adding heat.

This activity was observed by a Qualified Nuclear Engineer who expressed some concerns but failed to adequately communicate technical advice for excessive control rod manipulation to shift management. Operations supervision was so focused on pump restoration activities that appropriate oversight of control rod manipulations was not provided. During the time period when the operator excessively manipulated control rods, 39 people were in the control room envelope, with 15 people in the immediate vicinity of the areas where the operator and unit supervisor were stationed.

This event demonstrated that administrative controls not only failed to prevent a mistake, they also failed to correct the mistake in an authorized manner.

Zion Units 1 and 2 and Indian Point Unit 3 : (NRC Information Notice 94-75, Minimum Temperature for Criticality (October 14, 1994))[46]

The NRC reported that analysis performed by Westinghouse for Zion assumed a nominal hot-zero-power operating temperature of 547 degrees F, but Zion's Tech Specs allowed reactor criticality if the average reactor coolant system temperature was greater than 500 degrees F. The safety analysis for Indian Point 3 was also performed at 547 degrees F, but its Tech Specs allowed criticality at 450 degrees F. A review of Indian Point records indicated that the reactor was brought critical at temperatures below 547 degrees F several times before 1988. The lowest temperature during these instances was 521 degrees F.

The NRC noted that reactor criticality at temperatures below the analyzed conditions had several adverse safety implications, including making the power range ex-core instrumentation indicate non-conservatively and making the moderator temperature coefficient more positive.

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46 A copy of this document is attached as Exhibit B-44.

This report shows that compliance with a Tech Spec requirement provides adequate safety margins only when the Tech Spec limit is bounded by the plant's safety analyses.

Appendix C

Assessing the Probability and Consequences of Criticality Events in Fuel Pools

1. Introduction

This appendix provides technical background on the potential for inadvertent criticality in a fuel pool. Specifically, this appendix describes the steps that must be taken to assess the probability and consequences of a criticality event, and sets forth some interim findings about the Millstone Unit 3 pool. These findings are necessarily of an interim nature, because CCAM/CAM has not identified any systematic assessment of the probability and consequences of a pool criticality. Neither the NRC Staff nor the nuclear industry has attempted such an assessment or compiled the record of experience and other factual data that would support an assessment.

The probability of a criticality event is discussed here in terms of six steps. First, the various types of criticality scenario are identified. Second, the probability of these scenarios is explored from a qualitative perspective. Third, the process of determining the envelope of criticality in a pool is described. Fourth, the potential for fuel mispositioning is outlined, drawing upon actual experience. Fifth, the potential for a reduced concentration of soluble boron is outlined, again drawing upon experience. Sixth, available criticality calculations that apply to the Millstone Unit 3 pool are summarized, thereby showing the broad outlines of the envelope of criticality for this pool.

Then, the nature and consequences of a criticality event are discussed. Finally, some conclusions are presented.

2. Probability of a Criticality Event

2.1 Overview

Analytic techniques are available for assessing both the probability and consequences of a criticality event in a fuel pool. For example, relevant techniques have been developed for probabilistic risk assessments (PRAs) at nuclear power plants. However, CCAM/CAM has not identified any attempt, either by the NRC Staff, the nuclear industry or any other body, to conduct a systematic assessment of the probability and consequences of a pool criticality. Moreover, there has been no systematic effort by the NRC Staff or the nuclear

industry to compile the factual data that would be needed to support such an assessment. The relevant data would be drawn from actual operating experience at nuclear facilities.

In the absence of a systematic investigation, one can make only qualitative statements about the probability of a criticality event in a fuel pool, drawing from publicly available information.

2.2 Types of Criticality Scenario

This discussion focusses on the potential for a criticality event under abnormal conditions. Thus, for the purposes of this discussion, we ignore the possibility that a criticality event will occur in a fuel pool under normal conditions. In other words, if the pool contains as-specified fuel in as-specified fuel storage racks, and other parameters such as water temperature and soluble boron concentration are within their specified range, then we assume that a subcritical margin of reactivity will exist.

Nevertheless, criticality could occur under normal conditions if there is a major error in the calculations that are performed to support the design and installation of the fuel storage racks. Appendix B shows that errors have occurred in calculations of this kind. For example, at Braidwood Unit 1, an incorrect assumption about the location of Boral panels was carried forward through successive calculations from 1987 to 1997. Also, at Millstone Unit 2, new calculations showed a Keffective of 0.963 whereas previous calculations, which had employed two inappropriate assumptions, showed a Keffective of 0.922. That is a substantial error, in a non-conservative direction. The potential for errors of this type is smallest when the rack design relies solely on geometry (the center-center distance between fuel assemblies) to prevent criticality.

Under abnormal conditions, a variety of scenarios could lead to inadvertent criticality in a fuel pool. The number of potential scenarios is greater when a greater number of means are used to suppress criticality.

If the prevention of criticality in the pool under normal conditions relies entirely on the use of geometrically safe racks, then three types of scenario could lead to criticality under abnormal conditions. First, an earthquake, drop of a heavy object into the pool or other mechanical insult might alter the rack geometry sufficiently to cause criticality. Second, fuel assemblies that are more reactive (e.g., with a higher-than-specified enrichment in U-235) than the specified limit

for fresh fuel entering this facility might be placed in the racks. Third, fuel assemblies might be placed inside or outside a rack in a manner that does not conform to the intended geometry of fuel placement.

If the prevention of criticality under normal conditions relies not only on rack geometry but also on the neutron-absorbing properties of the racks, then the three types of scenario outlined above could lead to criticality. In addition, criticality might arise if neutron-absorbing material is displaced from its intended position (e.g., if Boral panels become detached from the racks).

If the prevention of criticality under normal conditions relies not only on rack geometry and the neutron-absorbing properties of the racks, but also on restricted fuel burnup/enrichment or age, or on the presence of soluble boron, then criticality could arise through one of the scenarios outlined above or through additional scenarios. These additional scenarios would involve mispositioning of fuel assemblies, a reduction in the concentration of soluble boron in the pool water, or a combination of these occurrences. In this context, "mispositioning" would involve the placement in a rack of one or more fuel assemblies whose burnup/enrichment or age is not within the specified range. In scenarios that combine fuel mispositioning with a reduced concentration of soluble boron, the mispositioning could either precede or follow the reduction in boron concentration.

2.3 Scenario Probability from a Qualitative Perspective

Some of the criticality scenarios outlined in Section 2.2 would involve significant mechanical insult (e.g., an earthquake that disrupts the geometry of a rack) or mechanical failure (e.g., the detachment of Boral panels from racks). If the pool and the racks are designed, built and operated to prevailing standards, these scenarios will have a relatively low probability.

Another type of criticality scenario involves the placement of fuel assemblies inside or outside a rack in a manner that does not conform to the intended geometry of fuel placement. For example, a fuel assembly might be dropped and come to rest in a horizontal position across the top of a rack, or in a vertical position between racks. The possible configurations of this kind are limited by the arrangement of the racks and the practice of moving fuel assemblies one at a time. Thus, this type of criticality scenario will also have a relatively low probability.

The remaining types of criticality scenario involve failures of administrative controls. One scenario involves the placement in a rack of fuel that is more reactive (e.g., with a higher enrichment in U-235) than the level specified for fresh fuel entering this facility. Facility licensees, and their contractors and vendors, seek to prevent such an event by employing administrative controls of a "one-time" variety. For example, the level of U-235 enrichment of a fresh fuel assembly will be verified at several points in the manufacturing process. Occurrence of a criticality would be attributable to failure of the one-time administrative controls either during fuel fabrication or fuel delivery. This type of criticality scenario will have a relatively low probability, because one-time administrative controls have a relatively low likelihood of failure.

In other criticality scenarios that involve failures of administrative controls, the failed controls will generally be of the "ongoing" variety. In particular, if restrictions on fuel burnup/enrichment or age, or the presence of soluble boron, are exploited as means of criticality suppression under normal conditions, the implementation of those means will rely upon ongoing administrative controls. Failure of those administrative controls could lead to criticality scenarios that involve the placement in a rack of fuel assemblies with inappropriate burnup/enrichment or age, a reduction in the concentration of soluble boron in the pool water, or a combination of these occurrences.

Over time, ongoing administrative controls will have a much higher cumulative probability of failure than one-time controls. Thus, criticality scenarios that involve fuel mispositioning (the placement in a rack of fuel assemblies with inappropriate burnup/enrichment or age), a reduction in the concentration of soluble boron in the pool water, or a combination of these occurrences, will have a much higher probability than other criticality scenarios. In illustration, CCAM/CAM concludes from the historical record presented in Appendix B that fuel mispositioning is a likely event.

2.4 Determining the Envelope of Criticality in a Pool

An important step in understanding the potential for criticality in a pool is to determine the range of conditions in which criticality will occur. The boundary of this range constitutes the envelope of criticality in the pool. A determination of the envelope is a necessary precursor to a systematic assessment of the probability of a criticality event, and must also precede an application of the Double Contingency Principle (as described in Draft Reg. Guide 1.13).

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To illustrate the concept of an envelope of criticality, consider the set of criticality scenarios that involve fuel mispositioning (the placement in a rack of fuel assemblies with inappropriate burnup/enrichment or age), a reduction in the concentration of soluble boron in the pool water, or a combination of these occurrences. In order to determine the envelope of criticality for these scenarios, one would begin by specifying a particular pool and rack configuration, and the most reactive fuel assembly that could be placed in the pool (this may be a fresh fuel assembly). Next, one would identify the possible range of fuel mispositioning events. Then, one would determine the combinations of fuel mispositioning events and soluble boron concentrations that will yield a Keffective of exactly 1 (or, if a factor of safety is used, some lesser value of Keffective such as 0.95). The set of these combinations would be the envelope of criticality in the pool, for these scenarios.

Discovery in a previous case suggests that no entity in the United States has undertaken the calculations necessary to determine the envelope of criticality in a fuel pool.[1] During depositions of NRC Staff witness Dr Laurence Kopp and CP&L witness Dr Stanley Turner, Orange County's attorney asked these witnesses how they would determine the envelope of criticality in a fuel pool, as defined above. Both witnesses' responses indicated that neither the NRC Staff, CP&L nor CP&L's contractor Holtec has given significant attention to developing a thorough understanding of the potential for criticality scenarios of the type discussed here.

2.5 The Potential for Mispositioning of Fuel

Appendix B reviews the record of fuel mispositioning at US nuclear power plants, drawing from documents that are currently available to CCAM/CAM. These documents almost certainly do not reveal the full historical record of relevant events, for reasons that are explained in Appendix B. Nevertheless, Appendix B shows that fuel mispositioning, involving placement in a fuel pool of one or more fuel assemblies with inappropriate burnup/enrichment or age, is a likely occurrence.

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[1] Detailed Summary of Facts, Data and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant, with Respect to Criticality Prevention Issues (Contention TC-2), January 4, 2000 (see Appendix C, page C-5).

Many of the relevant events described in Appendix B directly involved the mispositioning of one or more fuel assemblies in a fuel pool. The other relevant events included fuel handling errors that affected a reactor core, or fuel handling errors that occurred in a fuel pool but did not directly lead to a mispositioning of fuel. These other events are relevant because they show that ongoing administrative controls related to fuel handling and management are likely to fail. This information supports our finding that fuel mispositioning in a pool is a likely occurrence.

The fuel mispositioning events described in Appendix B included events where more than one fuel assembly was mispositioned. Notably, at Oyster Creek, up to 184 fresh fuel assemblies were inappropriately stored in the spent fuel pool. Oyster Creek's safety analysis had not considered the possibility that fresh fuel would be stored in the pool. Some of the mispositioning events described in Appendix B involved only one fuel assembly but could have involved multiple assemblies, because these events were attributable to failures in administrative controls that governed many assemblies.

2.6 The Potential for a Reduced Concentration of Soluble Boron

The concentration of soluble boron in the water in a fuel pool will be reduced if water with a lower concentration of soluble boron is added. At a typical PWR nuclear plant, the additional water could come from a variety of unborated water sources that interface with the fuel pool, including: the component cooling water system (which removes heat from the fuel pool heat exchangers); the demineralizer system (which is used to sluice and refill the demineralizer); the reactor makeup system (which provides makeup for evaporation losses in the fuel pool); the fire protection system; and the service water system.[2]

In addition, where a fuel pool and other water-containing compartments (e.g., fuel transfer canal, cask pit, reactor refuelling cavity) are interconnected but are separated by removable gates, as is true at Millstone Unit 3, water from one compartment could mix with water from another compartment if a gate is removed. If one compartment has a lower concentration of soluble boron, the mixing process will reduce the concentration in the other compartment.

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[2] Westinghouse Electric Corp, "Westinghouse Owners Group Evaluation of the Potential for Diluting PWR Spent Fuel Pools", WCAP-14181, July 1995, page 2-7.

Other soluble boron dilution scenarios can be postulated or have occurred. In illustration, in July 1994 the soluble boron concentration in the McGuire Unit 1 pool was inadvertently reduced from 2,105 ppm to 1,957 ppm (a 7 percent reduction). This event is summarized in Appendix B. Unborated water that was used to decontaminate a drained fuel transfer canal was transferred by a submersible pump to the fuel pool.

A study by the Westinghouse Corporation sought to estimate the probability of soluble boron dilution at PWR plants.[3] This study examined a generic, "composite" plant. It sought to estimate the probability of diluting the soluble boron concentration in the fuel pool from 2,200 ppm to 1,380 ppm (a 37 percent reduction), yielding a probability estimate of 3.8×10^{-7} per reactor-year. The study did not summarize the historical record of relevant events, such as the July 1994 event at McGuire Unit 1. Nor did this study examine mixing among pools, transfer canals and the refuelling cavity in situations when these volumes have previously been separated by gates. In addition, this study was performed by an interested party (Westinghouse). According to the NRC Staff's expert, Dr. Laurence Kopp, the report was never reviewed by the NRC Staff, because the Staff considered that a generic study would not be very valuable in light of the great variation among nuclear plants with respect to such factors as the volume of water that can be inserted into a pool for dilution, the mode of inserting it, and the capacity of the pools.[4] Thus, the study's estimate of the probability of soluble boron dilution should be viewed as a lower bound, and not as a reliable estimate.

As explained in Section IV of the Summary to which this appendix is attached, the technical analysis made available in this proceeding was insufficient to establish the potential for dilution of soluble boron at the Millstone Unit 3 pool.

2.7 Criticality Calculations for the Millstone Unit 3 Pool

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[3] WCAP-1418, Westinghouse Owners Group, Evaluation of the Potential for Diluting PWR Spent Fuel Pools (July 1995).

[4] Deposition of Dr. Laurence I. Kopp in a previous proceeding (Docket No. 50-400-LA, ASLBP No. 99-762-02-LA), Tr. at 36-39. A copy of the relevant pages of Dr. Kopp's deposition is attached as Exhibit C-1.

NNECO has provided the results of some calculations related to criticality.[5] When supplemented by the results of criticality calculations performed by the NRC Staff in a previous proceeding, these results illustrate the envelope of criticality for the Millstone Unit 3 pool, for scenarios involving fuel mispositioning and the dilution of soluble boron. These results are not, however, adequate for an accurate determination of the envelope.

Some relevant calculations were performed by the NRC Staff in connection with a previous proceeding (Docket No. 50-400-LA, ASLBP No. 99-762-02-LA), and were reported in an internal NRC Staff memorandum of November 5, 1999 from Tony Ulses to Ralph Caruso.[6] This document is hereafter described as the "Ulses Memorandum". The calculations assumed a fuel mispositioning event in which an entire PWR rack of the type proposed for pools C and D at the Harris plant is loaded with fresh PWR fuel assemblies enriched 5 wt% in U-235. The proposed Region 2 racks in the Millstone Unit 3 pool would be similar to the PWR racks proposed for pools C and D at Harris, and the fuel would be similar. Thus, the calculational results described in the Ulses Memorandum provide a rough approximation to the results that one would expect for a similar calculation for Region 2 of the Millstone Unit 3 pool.

The SCALE modular code system was used by the NRC Staff for the calculations described in the Ulses Memorandum. The Memorandum reported its results in terms of a neutron multiplication factor (designated hereafter as K), without discriminating between Kinfinite and Keffective. Assuming an absence of soluble boron and a rack loaded with fresh fuel of 5% enrichment, the SCALE calculations yielded a K of 1.19378.

Table 2 of Exhibit C-2 provides estimates of Keffective for placement of fresh fuel of 5% enrichment in the proposed Region 2 of the Millstone Unit 3 pool. For a soluble boron concentration of 2,000 ppm, a rack filled with fresh fuel would exhibit a Keffective of 0.9842, which is close to criticality. For a boron concentration of 800 ppm, eight fresh fuel assemblies in an otherwise empty rack would exhibit a Keffective of 0.9794, while five fresh assemblies in an otherwise

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[5] NNECO's Supplementary Response to CCAM/CAM's Third Set of Interrogatories, June 21, 2000. Tables 1-3 of that Response are attached herewith as Exhibit C-2.

[6] A copy of this document is provided herewith as Exhibit C-3.

filled rack would exhibit a Keffective of 0.9663.[7] Both situations are close to criticality.

If the findings of the Ulses Memorandum are assumed to be applicable to the proposed Region 2 of the Millstone Unit 3 pool, one finds that, for an absence of soluble boron, a rack filled with fresh fuel of 5% enrichment would exhibit a Keffective of about 1.2. In this situation, the rack would be supercritical on prompt neutrons alone, an outcome whose implications are described below.

To summarize, one can see that the envelope of criticality in the proposed Region 2 of the Millstone Unit 3 pool would include a point where a rack is filled with fresh, 5%-enriched fuel and the soluble boron concentration is somewhat below 2,000 ppm. Another point on the envelope would combine a boron concentration of 800 ppm with the unaccompanied presence of somewhat more than eight fresh, 5%-enriched assemblies. A third point on the envelope would combine an absence of boron with the presence of a small number (more than one) of fresh, 5%-enriched fuel.

Table 1 of Exhibit C-2 shows that criticality would not occur in Region 1 (presumably in a 3-out-of-4 configuration) for any combination of soluble boron concentration and placement of fresh, 5%-enriched fuel (assuming that the racks retain their structural and material integrity). Table 3 shows that, for Region 3, the envelope of criticality would be somewhat like the envelope for Region 2.

3. Nature and Consequences of a Criticality Event

The major determinant of the consequences of a criticality event will be the cumulative energy release during the event. In turn, the cumulative energy release will be determined by several factors, including the rapidity with which a critical configuration is assembled, and the manner in which the system responds when fission energy is released.

Consider scenarios in which criticality occurs in the Milstone Unit 3 pool as a result of the mispositioning of fresh or high-reactivity fuel, combined with a reduced concentration of soluble boron. In such a scenario, the threshold of criticality could be crossed in either of two ways. First, the threshold could be

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[7] The age and burnup of the non-fresh fuel in the rack in the second case are not stated in Table 2 of Exhibit C-2, but are presumably within the proposed Tech Spec limits.

crossed while a fuel assembly with greater-than-specified reactivity is being placed in a rack that is already close to criticality because of previous fuel mispositioning combined with a previously reduced concentration of soluble boron. Second, the threshold could be crossed while soluble boron concentration is declining in a pool that is already close to criticality because of previous fuel mispositioning.

In both cases, the threshold of criticality would be crossed relatively slowly. However, the above-summarized calculations by NNECO and the NRC staff show that the final configuration could be critical on prompt neutrons alone. For example, NNECO finds that a close-to-critical configuration would exist in Region 2 (Keffective is 0.97) if five fresh assemblies are present in an otherwise full rack and the soluble boron concentration is 800 ppm. The completed placement of additional fresh assemblies in nearby locations and/or a reduction in the concentration of soluble boron could yield a Keffective of, for example, 1.01. That configuration would be critical on prompt neutrons alone, because the delayed neutron fraction for U-235 fission is 0.0065. The process of assembling such a configuration is discussed in later paragraphs of this Section.

In a situation of prompt-neutron criticality, the rate of fission would rise rapidly. The time between each generation of fission in a chain reaction could be about 10^{-4} seconds, in which case 1,000 generations of fission would occur in 0.1 seconds and 5,000 generations would occur in 0.5 seconds. If a Keffective of 1.01 were achieved for prompt neutrons alone (i.e., a Keffective of 1.0165 for all neutrons), then one fission in the first generation would lead to 2.1×10^{4} fissions at 0.1 seconds (during the 1,000th generation) and 4.0×10^{21} fissions at 0.5 seconds (during the 5,000th generation). Since one fission of U-235 releases about 200 MeV (3.2×10^{-11} Joules) of energy, the 5,000th generation of fission would release about 130 billion Joules of energy. This energy release would occur over a period of about 10^{-4} seconds, and would involve the burning of about 1.6 grams of U-235. For comparison, note that fission in a typical commercial nuclear reactor with a thermal power capacity of 3,000 MW will release, when the reactor is at full power, 3 billion Joules of energy per second.[8]

Clearly, a fuel pool criticality event of this kind would be self-limiting, and would not proceed to the point where 130 billion Joules of energy is released in

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[8] For background on this paragraph and the preceding paragraph, see: Anthony V Nero, "A Guidebook to Nuclear Reactors", University of California Press, 1979.

one generation of fission. The reactivity coefficients of this system are negative. Notably, a substantial energy release would lead to local boiling of the pool water, which would reduce reactivity. A cyclic process might occur, involving repeated episodes of local boiling. If initiated, such a cycle could continue until terminated by depletion of fissile material in the fuel, evaporation of water, or the addition of soluble boron to the pool.

Although a criticality event would be self-limiting, the energy release could be sufficient to damage the fuel. If damaged, the fuel could release radioactive material into the atmosphere of the pool building and from there to the external environment. Also, personnel in the pool building could be exposed to direct gamma and neutron radiation released during fission.

Let us turn again to the initial phase of the criticality, which was briefly addressed in earlier paragraphs in this Section. For the scenarios assumed here, the threshold of criticality would be crossed relatively slowly, either during placement of a fuel assembly or during a decline in the concentration of soluble boron. An interval of time, lasting from seconds to minutes or longer, would occur between the crossing of the threshold and the attainment of the maximally reactive configuration. During that time interval, the reactivity of the system would initially rise but would then be constrained by feedback mechanisms. A cyclic process might occur, in which reactivity repeatedly rises and falls, with a continuing rise in the peak reactivity until the maximally reactive configuration is reached. An alternative possibility is that the criticality event might self-terminate because the initial energy release destroys the critical configuration. For example, local boiling in a rack cell might expel a fuel assembly that is being lowered into the cell, thereby terminating the event.

The entire process of a hypothesized criticality event could be systematically analyzed, using known techniques such as those employed by PRA practitioners. No such analysis has been performed to date, so there is no analytic basis to estimate the potential radioactive release to the environment or the radiation dose within the pool building. Our scoping calculations show, however, that substantial reserves of energy are available for release during a criticality event. Thus, significant onsite and offsite radiation exposures are potential outcomes of a criticality event.

4. Conclusions

Criticality could occur in a fuel pool through various types of scenario. If criticality prevention relies solely on rack geometry and the presence of solid boron, some scenarios would involve the failure of administrative controls, but these controls would be of the one-time variety.

The exploitation of fuel burnup/enrichment or age, or the presence of soluble boron, as additional means of criticality control introduces additional criticality scenarios. These additional scenarios involve fuel mispositioning or soluble boron dilution, or combinations of these occurrences. Fuel mispositioning or the dilution of soluble boron will occur as a result of the failure of ongoing administrative controls.

The probability and consequences of a criticality event in a fuel pool could be systematically investigated, but this has not been done. From a qualitative perspective, it is clear that the scenarios which involve the failure of ongoing administrative controls have a much higher probability than the other scenarios.

Experience at US nuclear plants shows that fuel mispositioning, involving placement in a pool of one or more fuel assemblies with inappropriate burnup/enrichment or age, is a likely occurrence. Up to 184 fresh fuel assemblies have been inappropriately placed in a pool.

Experience also shows that the concentration of soluble boron in a pool can fall below specified levels. A variety of scenarios could yield substantial reductions in soluble boron concentration.

Calculations performed by NNECO and the NRC staff show that supercritical configurations could occur in the proposed Regions 2 and 3 of the Millstone Unit 3 pool if fresh fuel assemblies are mispositioned and the concentration of soluble boron is reduced. Some of these configurations would be critical for prompt neutrons alone, leading to the rapid release of potentially large amounts of energy.

Significant onsite and offsite radiation exposures are potential outcomes of a criticality event.

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Exhibit C-1

Internal NRC Staff memorandum from Tony Ulses to Ralph Caruso, November 5, 1999 ("Ulses Memorandum")

November 5, 1999

MEMORANDUM TO: Ralph Caruso, Chief
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Tony P. Ulses, Nuclear Engineer /s/
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: COMPLETION OF CRITICALITY ASSESSMENT OF MISLOADING ERROR IN HARRIS C AND D SPENT FUEL POOL

I have completed the analysis evaluating the potential for criticality from a misloading error if Shearon Harris begins to use high density storage racks in the currently inactive C and D spent fuel pools. The analysis discussed in the enclosed report assumes a worst case misloading error in which the entire rack is misloaded with fresh 5 w/o enriched Westinghouse 15x15 fuel which has been previously determined to be the most reactive PWR fuel type which could be loaded into the Harris pools. This analysis demonstrates that the multiplication factor will remain less than one (i.e. subcritical) for this postulated worst case scenario. The calculated eigenvalues are taken at upper 95/95 level and a manufacturing uncertainty of 1 percent has been added to the predicted value.

Enclosures:
As stated

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Evaluation of Postulated Worst Case Misloading Error for Harris C and D Spent Fuel Pools

Tony P. Ulses
November 2, 1999

1 Introduction

Carolina Power and Light (CP&L), the operator of the Shearon Harris nuclear power plant, requested a license amendment to activate the two unused spent fuel pools at the Harris site. The proposal is to use a "high density" storage configuration which requires the use of burnup credit racks. In the context of this report burnup credit racks refer to storage racks which require that the fuel has reached a pre-specified minimum burnup before it can be safely stored. The need for this burnup requirement is dictated by the fact that the inter-assembly spacing is reduced to achieve the desired "high density" configuration. Whenever one relies on a physical process such as burnup one needs to assess the impact of an assembly being inserted into the rack that has not reached the minimum acceptable burnup. Therefore, criticality analyses have been performed to assess the effect of an assembly misloading error in the Harris "C" or "D" spent fuel pool. In this analyses it was assumed that the entire rack was misloaded with UO_2 fuel enriched to 5 w/o U^{235} which is the highest enrichment allowed at commercial power plant's in the US. This would be the worst possible configuration.

2 Definition of Problem

In this analyses we will assess the impact of a worst case misloading accident by predicting the multiplication factor of the system. To this end, we will perform three base analyses and one sensitivity calculation. Two of the base analyses are intended to assess the staff's criticality calculations against the licensee calculations and the final analyses will assess the worst case misloading accident. The two comparative calculations are important because they will allow an assessment of the licensee method's and will serve to strengthen the staff's position with respect to these methods. A brief description of the problems will follow:

Typical Parameters

Fuel type:	Westinghouse 15x15 Assembly Enriched to 5 w/o U^{235}
Rack type:	Holtec High Density
Boundary Conditions:	Reflective in x, y, and z
# of Histories:	1000 groups of 3000 particles for a total of 3 million histories

Problem 1

This problem is extracted from reference 1. The rack should be assumed to be loaded with fresh fuel without soluble boron. All dimensions should be nominal.

Problem 2

This problem is the licensing basis for the storage racks. The rack should be loaded with fuel burned to 41.7 Mw$_d$/KgU. The depletion is to be performed assuming three cycles of operation with an average boron concentration of 900 ppm, a specific power of 42 kW/KgU, nominal fuel and clad temperature and slightly higher than expected moderator temperature. The criticality analyses should assume no soluble boron is present and credit will be taken for actinides and fission products. All dimensions should be nominal.

Problem 3

This problem assesses the effect of the worst case misloading accident. The rack should be loaded with fresh fuel and one should assume that the soluble boron is present. All dimensions should be nominal.

3 Description of Methods

The SCALE (ref. 2) system was chosen for both the criticality analyses and the burnup calculations. The SCALE system has been extensively assessed and validated for these types of calculations (refs. 3 - 5). The SAS2H sequence was used for the depletion calculations and the CSAS6 sequence was used for the criticality calculations. Both of these sequences use BONAMI and NITAWL-II to process cross sections into a problem specific AMPX working library. SAS2H uses XSDRN and ORIGEN to deplete the fuel and CSAS6 uses KENO-VI for criticality calculations. Both the 44 group and the 238 group ENDF/B-V based AMPX libraries were used in the criticality analyses and the 44 group AMPX library was used for depletion.

4 Presentation and Discussion of Results

The results for problems 1 and 2 are presented in table 1. For comparative purposes, we have included the results from the licensee's contractor (ref. 1). This comparison reveals that the licensee method seems to predict slightly higher mulitplication factors (as much as 2% overall). However, given the differences in the methods the staff considers this to be excellent agreement and this gives us a great deal of confidence in the methods being used by both the staff and the licensee.

Table 1 Comparison of Results for Problem 1 and Problem 2

	CASMO	MCNP	SCALE[1]
Problem 1	1.2076	1.2056	1.19378
Problem 2	0.9126	N/A	0.8940

[1]The SCALE results are the staff calculation.

The multiplication factor predicted for problem 3 is 0.978 at the upper 95/95 interval using the 44 group library and 0.979 using the 238 group library. The 238 group library was also used for this problem to ensure that collapsing spectrum used to generate the 44 group library from the 238 group library did not introduce any significant bias into the results. This demonstrates that even assuming the worst case misloading error (i.e. misloading an entire rack with fresh fuel) the rack will remain subcritical when one considers the soluble boron which will be present in the pool.

In order to assess the adequacy of multiplication factors predicted using Monte Carlo methods it is prudent to consider, in addition to the number of histories tracked, how well the spatial and energy domains of the problem were sampled. To this end, we have attached the spectrum

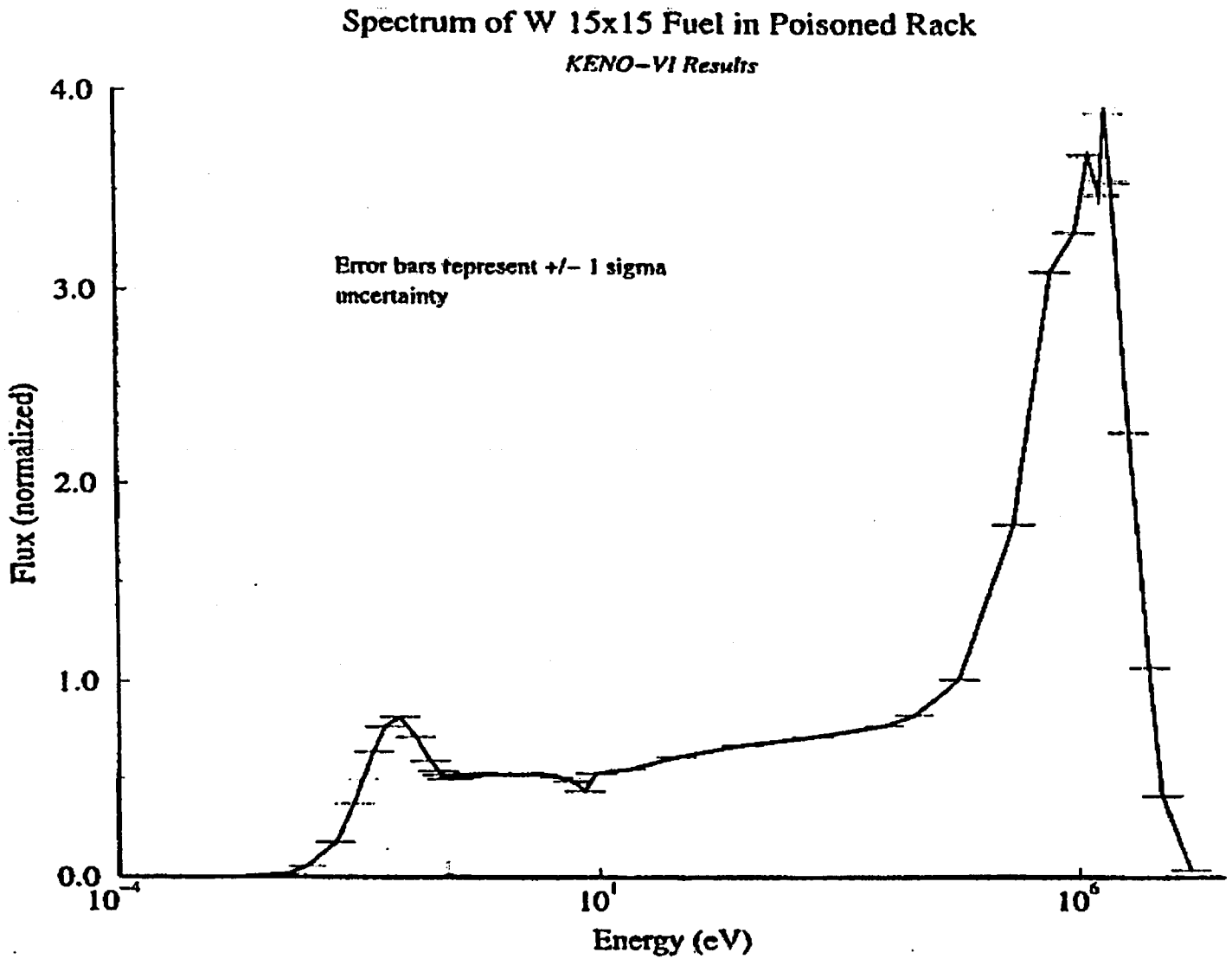

Figure 1 KENO Predicted Spectrum for W 15x15 Fuel Assembly

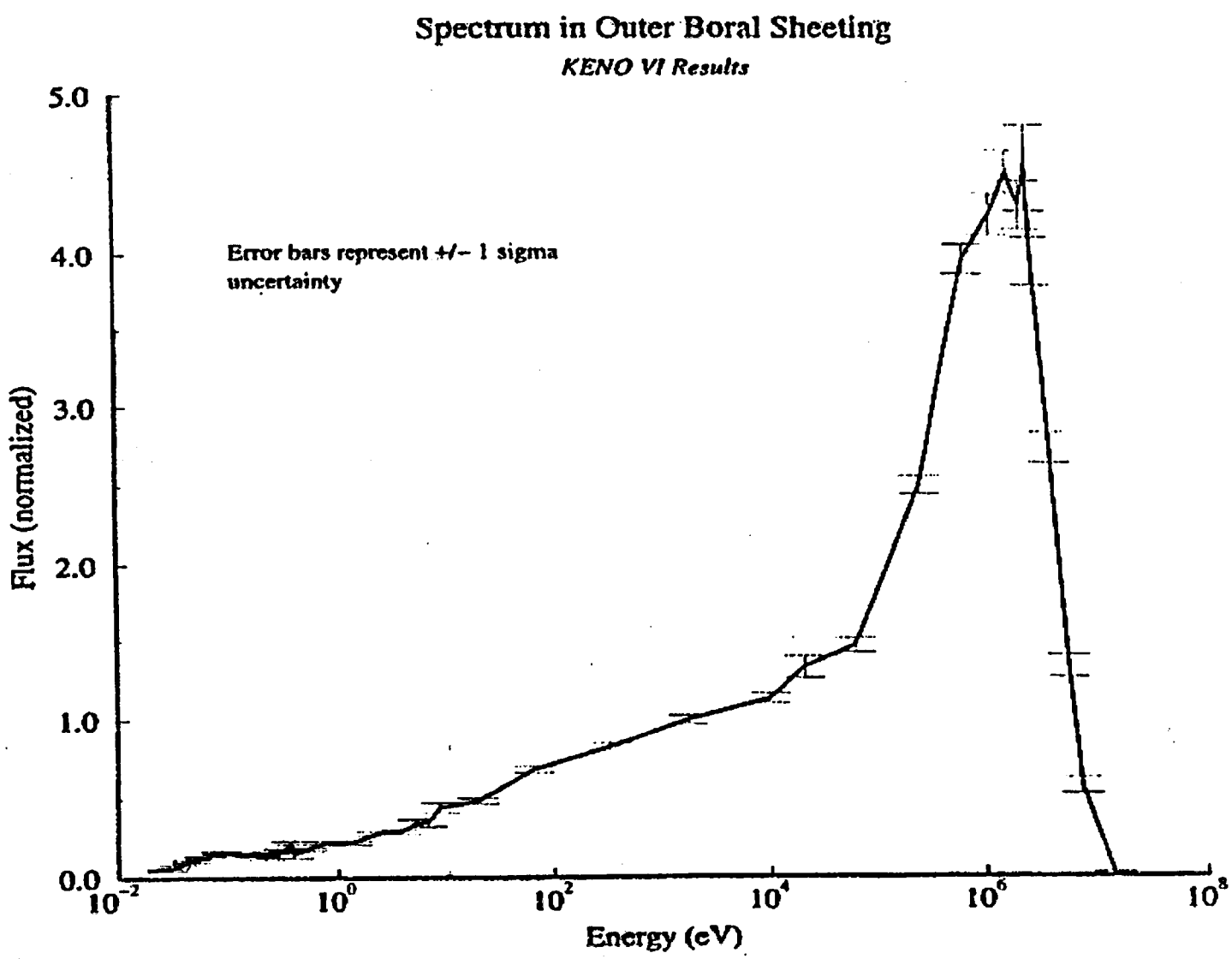

Figure 2 KENO Predicted Spectrum in Outer Boral Sheeting

Appendix A

Excerpt from KENO-VI Major Edit

1 keno-vi input for storage cell calc. for holtec rack w/ 15x15 w

0fluxes for global unit

0group	region 1 flux	region 1 percent deviation	region 2 flux	region 2 percent deviation	region 3 flux	region 3 percent deviation	region 4 flux	region 4 percent deviation	region 5 flux	region 5 percent deviation	region 6 flux	region 6 percent deviation
1	0.000E+00	0.00	1.376E-04	5.36	8.973E-06	18.11	1.932E-05	19.59	1.823E-05	18.58	2.150E-05	12.41
2	0.000E+00	0.00	4.190E-04	3.46	5.856E-05	8.68	5.653E-05	8.31	4.404E-05	9.44	4.438E-05	8.41
3	0.000E+00	0.00	1.267E-03	1.92	1.656E-04	5.23	1.544E-04	5.92	1.281E-04	5.35	1.475E-04	4.97
4	0.000E+00	0.00	4.204E-03	1.14	5.437E-04	3.46	5.072E-04	3.55	4.983E-04	3.51	4.957E-04	2.91
5	0.000E+00	0.00	2.834E-03	1.37	3.175E-04	3.97	3.393E-04	3.74	3.345E-04	4.01	3.336E-04	3.86
6	0.000E+00	0.00	8.974E-04	2.41	9.913E-05	5.97	1.171E-04	7.88	1.041E-04	6.83	1.040E-04	6.43
7	0.000E+00	0.00	3.574E-03	1.33	4.377E-04	3.59	4.251E-04	3.72	3.972E-04	4.34	4.402E-04	3.67
8	0.000E+00	0.00	4.386E-03	1.18	5.304E-04	3.18	5.120E-04	3.19	4.895E-04	3.44	5.272E-04	3.33
9	0.000E+00	0.00	6.307E-03	1.08	7.926E-04	3.15	7.232E-04	3.15	6.767E-04	3.19	7.520E-04	2.96
10	0.000E+00	0.00	1.103E-02	0.78	1.355E-03	2.44	1.291E-03	2.43	1.246E-03	2.54	1.271E-03	2.40
11	0.000E+00	0.00	1.178E-02	0.74	1.464E-03	2.26	1.336E-03	2.36	1.340E-03	2.54	1.391E-03	2.39
12	0.000E+00	0.00	7.178E-03	0.92	8.611E-04	3.00	8.099E-04	2.97	7.276E-04	3.03	7.950E-04	2.93
13	0.000E+00	0.00	1.595E-03	1.71	2.171E-04	5.22	1.834E-04	5.38	1.810E-04	5.70	1.772E-04	5.20
14	0.000E+00	0.00	7.130E-03	0.92	8.294E-04	2.75	7.465E-04	2.97	7.295E-04	3.31	8.029E-04	2.87
15	0.000E+00	0.00	6.261E-03	0.92	7.122E-04	2.72	6.722E-04	2.81	6.210E-04	2.98	6.581E-04	2.97
16	0.000E+00	0.00	5.505E-03	0.92	5.951E-04	2.66	5.580E-04	2.90	5.222E-04	2.90	5.567E-04	2.73
17	0.000E+00	0.00	3.273E-03	1.15	3.484E-04	3.02	3.119E-04	3.32	2.897E-04	3.31	3.040E-04	3.06
18	0.000E+00	0.00	2.444E-03	1.36	2.262E-04	3.42	2.102E-04	3.45	2.065E-04	3.42	2.197E-04	3.25
19	0.000E+00	0.00	4.374E-04	2.80	3.954E-05	7.46	3.452E-05	6.89	3.002E-05	7.71	3.751E-05	8.08
20	0.000E+00	0.00	5.568E-04	2.75	4.471E-05	6.49	4.308E-05	6.56	3.613E-05	6.99	4.229E-05	6.87
21	0.000E+00	0.00	4.168E-04	2.97	3.427E-05	6.93	2.959E-05	7.91	3.034E-05	7.48	3.174E-05	7.49
22	0.000E+00	0.00	7.767E-04	2.22	5.679E-05	5.29	6.221E-05	5.63	5.160E-05	5.67	5.866E-05	5.56
23	0.000E+00	0.00	8.810E-04	2.08	6.311E-05	4.74	6.017E-05	4.94	5.510E-05	4.97	6.113E-05	4.85
24	0.000E+00	0.00	9.433E-04	1.98	5.403E-05	5.21	5.279E-05	5.27	5.165E-05	5.02	5.716E-05	4.83
25	0.000E+00	0.00	7.081E-04	2.24	4.488E-05	5.09	3.932E-05	5.31	3.755E-05	5.45	3.815E-05	5.11
26	0.000E+00	0.00	6.778E-04	2.21	3.444E-05	5.51	3.357E-05	5.09	2.928E-05	5.60	3.339E-05	5.59
27	0.000E+00	0.00	8.796E-05	4.92	4.091E-06	13.38	5.436E-06	12.26	4.366E-06	15.39	4.106E-06	14.27
28	0.000E+00	0.00	9.516E-05	5.12	6.096E-06	12.92	4.348E-06	14.18	3.879E-06	14.43	4.893E-06	13.21
29	0.000E+00	0.00	1.080E-04	4.50	5.983E-06	13.01	5.313E-06	15.04	5.081E-06	12.86	6.356E-06	11.50
30	0.000E+00	0.00	2.454E-04	3.50	1.201E-05	8.66	1.019E-05	11.42	1.107E-05	8.96	1.019E-05	8.98
31	0.000E+00	0.00	1.288E-04	3.78	5.914E-06	11.15	6.818E-06	13.22	5.718E-06	12.02	5.699E-06	12.93
32	0.000E+00	0.00	1.513E-04	3.91	6.783E-06	10.57	6.677E-06	10.90	6.587E-06	11.27	7.080E-06	9.90
33	0.000E+00	0.00	1.739E-04	3.52	6.496E-06	9.67	6.806E-06	9.95	7.721E-06	10.56	6.509E-06	10.88
34	0.000E+00	0.00	4.281E-04	2.47	1.835E-05	6.12	1.563E-05	6.45	1.648E-05	6.63	1.735E-05	6.08
35	0.000E+00	0.00	6.916E-04	1.90	2.472E-05	5.58	2.395E-05	5.19	2.208E-05	5.36	2.412E-05	5.32
36	0.000E+00	0.00	6.888E-04	1.78	2.515E-05	4.84	2.490E-05	5.24	2.035E-05	6.30	2.428E-05	4.73
37	0.000E+00	0.00	5.795E-04	1.93	1.896E-05	5.63	1.813E-05	5.20	1.732E-05	5.52	1.871E-05	5.04
38	0.000E+00	0.00	3.240E-04	2.18	1.036E-05	7.29	8.607E-06	7.85	1.001E-05	7.77	9.824E-06	8.23
39	0.000E+00	0.00	3.261E-04	2.37	9.701E-06	8.07	8.411E-06	7.75	7.653E-06	7.96	9.549E-06	7.42
40	0.000E+00	0.00	1.468E-04	3.28	3.917E-06	10.32	4.053E-06	10.54	3.024E-06	12.74	3.141E-06	11.64
41	0.000E+00	0.00	3.566E-04	2.29	1.058E-05	6.57	9.020E-06	7.10	8.858E-06	7.00	9.430E-06	6.96
42	0.000E+00	0.00	3.604E-05	5.88	1.009E-06	27.70	8.304E-07	19.11	8.564E-07	25.72	8.251E-07	21.12
43	0.000E+00	0.00	3.968E-05	5.42	9.087E-07	18.02	1.018E-06	16.41	8.949E-07	30.82	6.629E-07	20.54
44	0.000E+00	0.00	6.744E-06	11.51	2.102E-07	37.98	1.685E-07	40.02	1.729E-08	70.77	2.139E-07	37.00

Exhibit C-2

NNECO's Supplementary Response to CCAM/CAM'S Third Set of Interrogatories, June 21, 2000 (Tables 1-3)

TABLE 1

Criticality Calculations for Region 1

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.7611	k_{eff} well below critical
800 Technical Specification limit	Completely filled with fresh fuel of 5% enrichment	0.8916	Remains subcritical at Technical Specification limit of 800 ppm
0 Highly unlikely Loss of all soluble boron	Completely filled with fresh fuel of 5% enrichment	0.9728	Remains subcritical with system filled with fuel of maximum reactivity and concurrent loss of all soluble boron

* The k-effective values do not include bias and manufacturing tolerances, which are usually about 0.015Δk in Region 1.

TABLE 2

Criticality Calculations for Region 2

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.9384	Multiple accident condition remains sub-critical
2,000 Boron dilution	Completely filled with fresh fuel of 5% enrichment	0.9842	Minimum Boron concentration of 2000 ppm Boron to assure sub-criticality for multiple accident scenario
800 Technical Specification limit	8 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 2 rack	0.9794	Multiple accident with 8 fresh fuel assemblies remains sub-critical at Technical Specification limit of 800 ppm Boron
800 Technical Specification limit	5 assemblies fresh fuel of 5% enrichment mis-loaded into Region 2 otherwise filled with spent fuel	0.9663	Multiple accident with 5 fresh fuel assemblies remains sub-critical at Technical Specification limit of 800 ppm Boron
0 Loss of all soluble Boron	3 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 2 rack	0.9241	Maximum number of concurrent accidents in otherwise empty Region 2 with loss of all soluble Boron
0 Loss of all soluble Boron	1 assembly fresh fuel of 5% enrichment accidentally mis-loaded into Region 2 otherwise filled with spent fuel	0.9450	Single misplaced assembly accident with concurrent loss of all soluble boron

* k-effective values do not include bias and manufacturing tolerances which are usually about 0.01Δk for fresh fuel (Cases 1, 2, 3, and 5 above). For Cases 4 and 6 above, with spent fuel assemblies present in the Region 2 racks, the bias and uncertainties could be as large as 0.019Δk.

TABLE 3

Criticality Calculations for Region 3

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.8503	Multiple accident condition – remains sub-critical
1,320 Boron dilution	Completely filled with fresh fuel of 5% enrichment	0.9811	Minimum soluble Boron concentration of 1,320 ppm to assure sub-criticality with multiple accident scenario
800 Technical Specification limit	8 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 3 rack	0.9752	Maximum number of concurrent accidents in Region 3 at the Technical Specification limit of 800 ppm Boron
800 Technical Specification limit	5 assemblies fresh fuel of 5% enrichment mis-loaded into Region 3 otherwise filled with spent fuel	0.9528	Maximum number of concurrent accidents in Region 3 at the Technical Specification limit of 800 ppm Boron
0 Loss of all soluble Boron	1 assembly of fresh fuel 5% enrichment mis-loaded into Region 3 otherwise filled with spent fuel	0.9707**	Single misplaced assembly of the maximum reactivity with concurrent loss of all soluble Boron

* k-effective values listed do not include bias and uncertainties which are about 0.018Δk for fresh fuel (Cases 1, 2, and 3 above) and 0.029% Δk when the racks are otherwise filled with spent fuel (Cases 4 and 5 above).

** A single misloaded assembly accident remains sub-critical at nominal spent fuel pool water temperatures, including bias and maximum uncertainties. However, because the temperature coefficient of reactivity is positive for Region 3, should a concurrent abnormal increase in pool temperatures occur, Region 3 could potentially reach a critical condition in the absence of all soluble boron. At 150°F, as little as 30 ppm of soluble boron would ensure sub-criticality, including bias and uncertainties.

Exhibit C-3

Deposition of Dr. Laurence I. Kopp in Docket No. 50-400-LA, ASLBP No. 99-762-02-LA (Transcript at 36-39)

1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of CAROLINA POWER & LIGHT COMPANY (Shearon Harris Nuclear Power Plant)	Docket No. 50-400-LA ASLBP No. 99-762-02-LA

DEPOSITION OF: LAURENCE I. KOPP, Ph.D.

DATE: November 4, 1999
Commencing at 2:15 p.m.

PLACE: Goya Conference Room
Four Points Sheraton Hotel
37611 U.S. Highway 19 North
Palm Harbor, Florida 34684

REPORTED BY: Dale E. DeFranco, RPR
Notary Public
State of Florida at large

ORIGINAL

1 APPEARANCES:

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1 that's why we decided this week to actually do a
2 calculation and see if would be true for Shearon Harris.
3 And we found we are subcritical for the entire rack.
4 Q. Okay. Under what circumstances, if any, and
5 under what regulatory requirements, if any, does the NRC
6 require the reporting of errors in controlling boron
7 concentration in the water of fuel storage pools?
8 A. I'm not sure if there would be any requirements
9 for reporting that. If the boron concentration were a
10 minimum boron concentration were in tec specs and if that
11 were violated during the surveillance interval, there
12 would be a certain amount of time where one could
13 reborate and get back up to the required minimum level.
14 And that would not be really I guess reportable unless
15 one did not borate in time. There's a certain interval
16 where you come back within regulations.
17 A. I see. And if you correct it with appropriate
18 intervals it's not a reportable event; is that what
19 you're saying?
20 A. Right.
21 Q. Okay. To the extent that boron dilution events
22 are reported to the NRC, does the NRC keep any
23 centralized record of boron dilution events that you
24 know?
25 A. It would be the same as the LER's for fuel

1 misplacements. There would be the LER's as far as I
2 know. We don't compile them but they're available.
3 Q. Has the NRC performed or obtained any analysis
4 or evaluation of nuclear power plant operator's
5 experience with controlling boron concentrations in fuel
6 storage pools?
7 A. Not that I know of.
8 MS. CURRAN: I'd like to ask the court reporter
9 to mark as Exhibit 10 an October 25th, 1996 letter
10 from Timothy E. Collins, Acting Chief, Reactor
11 System Branch, Division of System Safety and
12 Analysis, NRC, to Mr. Tom Green, Chairman
13 Westinghouse Owner's Group. Subject: Acceptance
14 for Referencing of Licensing Topical Report
15 WCAP-14416-P, Westinghouse Special Fuel Rack
16 Criticality Analysis Methodology.
17 Attached to this cover letter is a Safety
18 Evaluation by the Office of Nuclear Reactor
19 Regulation relating to Topical Report WCAP-14416-P.
20 (Whereupon, Exhibit Number 10 was
21 marked for identification.)
22 Q. Dr. Kopp, are you familiar with this document?
23 A. Yes, I am.
24 Q. If you would turn to page 10 -- actually page
25 10 is a continuation of a discussion that starts on page

1 8, Section 3.7 entitled Soluble Boron Credit Methodology;
2 isn't that correct?
3 A. Yes.
4 Q. If you look at the second full paragraph on
5 page 10 of the SER, I'd like to ask you about a sentence
6 that reads: "However, a boron dilution analysis will be
7 performed for each plant requesting soluble boron credit
8 to ensure that sufficient time is available to detect and
9 mitigate the dilution before the 0.95 k effective design
10 basis is exceeded and submitted to the NRC for review."
11 In parentheses, "Ref, dot, 29."
12 Can you explain to me what is meant by this
13 sentence and the reference to Ref 29?
14 A. Yes. This is the new methodology that I spoke
15 of earlier. This is one of the reasons for updating the
16 Grimes letter. This is a recent approval we gave for
17 crediting partial soluble boron in spent fuel pools. And
18 since we are allowing, not for Shearon Harris, but for
19 some reactors, credit for soluble boron under normal
20 conditions to meet .95, this would now require a new
21 accident to be evaluated which would be the boron
22 dilution event.
23 For other plants, such as Shearon Harris, which
24 do not take credit for soluble boron during normal
25 conditions, the fact that they calculate the five percent

2

1 subcriticality margin in pure water takes care of the
2 boron dilution event, that is complete dilution.
3 For these newer plants that want to take credit
4 for the new methodology. They still must show they are
5 subcritical with no boron, k effective is less than one,
6 but to meet the k arc criteria, k effective less than or
7 equal to .95, they can take credit for a certain amount
8 of soluble boron. So because of that we require them now
9 to do a boron solution analysis to show that they would
10 get them below .95 dilution event.
11 Q. Okay. But Reference 29 in parentheses, when I
12 turn to the back of this SER, Reference 29 is "Cassidy,
13 B., et. al., Westinghouse Owners Group Evaluation of the
14 Potential for Diluting PWR Spent Fuel Pools, WCAP-14181,
15 July 1995."
16 How does that Reference 29 relate to what we
17 were just reading on page 10?
18 A. That was a companion to this Westinghouse
19 report which requested credit for partial boron. In
20 order to prove that methodology I said they have to do a
21 boron dilution event analysis. And this other report
22 that you referenced shows how to do an analysis of a
23 boron dilution event in the PWR.
24 Q. So the reason for the mention of Reference 29
25 is that this is a way for licensees to do the boron

2

1 dilution analysis and that, that will meet NRC approval?
2 A. When they want credit for this methodology,
3 partial boron credit, yes.
4 Q. And has the NRC approved Reference 29 for that
5 purpose?
6 A. No. The approval of a boron dilution event we
7 decide is done on a case by case basis because the plans
8 vary so much. The amount of, the volume of water that
9 can be inserted into a pool for dilution varies from
10 plant to plant through the mode of inserting it, the
11 capacity of the pools vary. We decided a generic
12 dilution event would not be worth anything or worth much,
13 so we decided to, the people that wanted to accept this
14 methodology for partial boron credit would have to do a
15 plan specific for boron dilution analysis for their
16 specific spent fuel pool. That's why that boron dilution
17 event was never approved or accepted. It was a generic
18 type of topical report.
19 Q. Okay.
20 Q. Has the NRC performed or obtained any analysis
21 of the probability and/or consequences of potential
22 accidents resulting from improper boron concentration in
23 fuel storage pool water?
24 A. Only the analysis that shows that the zero PPM
25 of boron when there's still a five-percent subcritical

ERRATA SHEET

PLEASE ATTACH TO THE DEPOSITION.

IN THE CASE OF : Carolina Power & Light
CASE # : 99-7102-02 LA

Please read the transcript of your deposition and make note of any errors in the transcription on this page. Do NOT mark on the transcript itself. Please sign and date the transcript on PAGE _____. Please return both Errata Sheet and transcript to :

PAGE	LINE	ERROR/AMENDMENT	REASON FOR CHANGE
5	1	change "C." to "I."	wrong middle initial
7	15	change "1993" to "1983"	typo
8	17	change "cancel" to "canal"	"
11	11	change "simply" to "simplify"	"
22	3	change "additional" to "official"	"
22	10	change "concurrences" to "concurrences"	"
29	6	change "wasn't" to "was"	"
29	12	change "200" to "with 2000"	"
29	15	change "pucl" to "fuel"	"
38	6	change "are" to "effective"	"
38	9	change "solution" to "dilution"	"
39	7	change "plans" to "plants"	"
39	15	change "plan" to "plant"	"

CERTIFICATE OF DEPONENT

I, LAURENCE I. KOPP, do hereby certify that I have read the foregoing transcript of my deposition testimony and, with the exception of additions and corrections, if any, hereto, find it to be a true and accurate transcription thereof.

Laurence I. Kopp

12/27/99
DATE

Sworn and subscribed to before me, this the _____ day of _____, 19 _____.

_____
NOTARY PUBLIC IN AND FOR

My commission expires:

LIST OF EXHIBITS TO CCAM/CAM'S SUMMARY AND SWORN SUBMISSION

1. CR #M3-99-1148

2. Letter from Brian K. Grimes of the NRC Staff to All Power Reactor Licensees (April 14, 1978)

3. Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis) (December 1981)

4. Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants (August 19, 1998)

5. Reactor Engineering Logs, Millstone Unit 3, RFO 6

6. Millstone Unit 3 Refueling Outage No. 6 (1999): Excerpts from Reactor Engineering Logs

7. "The Daily Scorecard: Millstone Megawatts vs. Outage Barriers – All the facts, stats and at-bats for Unit 3's Refueling Outage" (May 1999)

8. Executive Summary, NNECO Nuclear Oversight Audit Report MP-3-99-A14 Refueling Activities (July 20, 1999)

9. CR-M3-2236

10. Transcript, Deposition of Michael C. Jensen, May 11, 2000

11. Matthew L. Wald, The New York Times, June 30, 2000, Page B1 ("Con Ed Put Off Plant Upgrade Over Rate Fear")

12. Memorandum of J.F. Beaupre (NNECO) to D.E. Anderson (NNECO) (June 24, 1999)

13. Letter of James C. Linville (NRC) to R.P. Necci (NNECO) (July 9, 1999)

14. Intervenors' Interrogatory A2 of Third Set of Interrogatories Directed to NNECO (May 18, 2000)

15. Set of photographs of Millstone Unit 3 Spent Fuel Pool provided by NNECO

16. McGuire Units 1 and 2: March 2, 2000 (Licensee Event Report 369/00/03) (March 30, 2000)

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

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

19. Byron Station: May 28, 1996 (Licensee Event Report 454/96-008-00) (June 25, 1996)

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

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

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

23. NNECO's Production of Documents Responsive to CCAM/CAM's First Set of Interrogatories and Request for Production (May 21, 2000), Interrogatory F-1 ("Please identify all instances of error (at Millstone or other nuclear plants) in managing, moving, placing or tracking fresh or spent fuel and all documents pertinent thereto.")

24. Reactor Engineering Log, Millstone Unit 3 RFO 5, pages 48-49

25. Transcript, Deposition of Robert Griffin, May 12, 2000

26. Transcript, Deposition of Anthony C. Attard, Ph.D., May 11, 2000

27. Transcript, Deposition of Laurence Kopp, Ph.D., May 11, 2000

28. NNECO's Supplementary Response to CCAM/CAM's Third Set of Interrogatories (June 21, 2000)

29. NRC Information Notice 94-13 (February 22, 1994)

30. AEC Press Release entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits")(November 22, 1965)

31. Internal AEC Memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966)(relevant excerpts)

32. Letter from J.J. DiNunno, AEC, to David Okrent, ACRS (October 25, 1966) and attached October 20, 1966 draft of General Design Criteria (relevant excerpts)

33. Letter from J.J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and Attached Draft of General Design Criteria (relevant excerpts)

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

35. Notice of Proposed Rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

36. Letter from William B. Cottrell, ORNL, to H.L. Price, AEC (September 6, 1967) and enclosed ORNL comments on proposed GDC

37. Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), enclosing General Design Criteria for Nuclear Power Plants (July 15, 1969)(relevant excerpts)

38. Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970), and enclosed letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970)

39. Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

EXHIBIT 1

CR M3-99-1148

Information Search on CR # M3-99-1148

Wednesday, June 14, 2000

CR No: M3-99-1148 CONTROL CIRCUIT FOR SPENT FUEL CASK CRANE DOES NOT PROVIDE THE TYPE OF SAFE OPERATION FOR WHICH IT WAS DESIGNED

Event Date:	Ops Screen Date:	Signif Level:	CR Status Code:	CR Process Code:	Reportability Code:
04/15/1999	04/15/1999	2	O	T	N

Document Notes:

Condition Description: The control circuit for the spent fuel cask crane (3MHF- CRN1) does not provide the type of safe operation for which it was designed. When placed in the creep mode, an interlock is bypassed which allows the crane to move while the overhead door is closed. In this configuration, it is possible to contact the door while operating the crane. Of even greater concern is that when the system is placed back into the normal mode, this interlock does not reset and still allows the crane to operate with the overhead door closed. The overhead door is a boundary between the spent fuel pool and the railway access and is normally closed except during crane operations. In addition to the condition above the crane restricted/unrestricted mode selector switch does not perform the function for which it was designed, nor does the TLS 3 switch which controls the heights that loads can be lifted. A review of the history of this crane indicates that this craneks control circuit has never performed as intended, and even during initial testing, numerous jumpers and bypass had to be installed to test the crane. Apparent Cause: The control circuit was not designed correctly to handle casks in accordance with the system description in the FASR. EWR M3-97052 was cancelled because plant management was attempting to reduce EWR backlog. CR M3-99-1148 was written because there is an urgent need to determine the size cask that can be used to move Unit 2 fuel to the Unit 3 fuel pool. Moving Unit 2 fuel to the Unit 3 pool is necessary to assure Unit 2 can operate to the end of its license. CA # 1 Initiate EWR M3-99-068 to investigate and resolve problems with spent fuel cask trolley control circuit. This action has been completed. CA # 2 Ensure implementation of EWR M3-99-068 to correct problems with spent fuel cask trolley or change corrective action plan.

AR Number:	Assign Nbr:	Assign Subject:	Assign Due Date:	Sched Reference	Mode Code:
99005924	01	M3-99-1148: LEAD: Control circuit for spent fuel cask crane	07/15/1999	NA	NA

Assign Status: COMPLETE Assign Type: CATI

Responsible Group: 3MGRTCHSUP

Assignment Text:

M3-99-1148, Level 2
This assignment was given to 3MGRTCHSUP as lead;
Reference CR M3-99-1148. Determine causal factors and provide corrective actions. CR has an open MRFF evaluation. If it is a functional failure, work with Tech Support Engineering and determine cause and develop corrective actions. If not a functional failure, include in CR closure package.
-
Condition Description:
The control circuit for the spent fuel cask crane (3MHF-CRN1) does not provide the type of safe operation for which it was designed. When placed in the creep mode, an interlock is bypassed which allows the crane to move while the overhead door is closed. In this configuration, it is possible to contact the door while operating the crane. Of even greater concern is that when the system is placed back into the normal mode, this interlock does not reset and still allows the crane to operate with the overhead door closed. The overhead door is a boundary between the spent fuel pool and the railway access and is normally closed except during crane operations. In addition to the condition above the crane restricted/unrestricted mode selector switch does not perform the function for which it was designed, nor does the TLS 3 switch which controls the heights that loads can be lifted. A review of the history of this crane indicates that this crane's control circuit has never performed as intended, and even during initial testing, numerous jumpers and bypass had to be installed to test the crane.
—
05/21/99: Please extend from 05/08/99 to 07/15/99, justification: Due to the Refueling Outage. Jim Beaupre/approved by Vince Spunar/eaf
—
5/22/99 Due date extended to 7/15/99 per request. lap

Assignment Completion Notes:
EWR M3-99-068 was initiated to evaluate and resolve the control circuit deficiencies. Jim Beaupre/am

Information Search on CR # M3-99-1148

Wednesday, June 14, 2000

AR Number:	Assign Nbr:	Assign Subject:	Assign Due Date:	Sched Reference	Mode Code:
99005924	02	M3-99-1148: MRFF EVALUATION	07/15/1999	NA	NA

Assign Status: COMPLETE **Assign Type:** CATI

Responsible Group: 3MGRTCHSUP

Assignment Text:

This assignment was given to 3MGRTCHSUP.
Reference CR M3-99-1148. Perform MRFF evaluation. If it is a functional failure, work with CR owner to determine cause and develop corrective actions. If not a functional failure, provide appropriate determination to CR owner for inclusion in CR closure package.
—
05/21/99: Please extend due date from 05/08/99 to 07/15/99, justification: Due to the Refueling Outage. Jim Beaupre/approved by Vince Spunar/eaf
—
5/22/99 Due date extended to 7/15/99 per request. lap

Assignment Completion Notes:

this condition is a design deficiency and is not a maintenance rule functional failure. Jim Beaupre/am

AR Number:	Assign Nbr:	Assign Subject:	Assign Due Date:	Sched Reference	Mode Code:
99005924	03	M3-99-1148: CA # 2	06/01/2001	NA	NA

Assign Status: ACCEPTED **Assign Type:** CATT

Responsible Group: MGRUNITSYS

Assignment Text:

Ensure implementation of EWR M3-99-068 to correct problems with spent fuel cask trolley or change corrective action plan

Assignment Completion Notes:

EXHIBIT 2

Letter from Brian K. Grimes of the NRC Staff to All Power Reactor Licensees (April 14, 1978)

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

April 14, 1978

To All Power Reactor Licensees

Gentlemen:

Enclosed for your information and possible future use is the NRC guidance on spent fuel pool modifications, entitled "Review and Acceptance of Spent Fuel Storage and Handling Applications". This document provides (1) additional guidance for the type and extent of information needed by the NRC Staff to perform the review of licensee proposed modifications of an operating reactor spent fuel storage pool and (2) the acceptance criteria to be used by the NRC Staff in authorizing such modifications. This includes the information needed to make the findings called for by the Commission in the Federal Register Notice dated September 16, 1975 (copy enclosed) with regard to authorization of fuel pool modifications prior to the completion of the Generic Environmental Impact Statement, "Handling and Storage of Spent Fuel from Light Water Nuclear Power Reactors".

The overall design objectives of a fuel storage facility at a reactor complex are governed by various Regulatory Guides, the Standard Review Plan (NUREG-75/087), and various industry standards. This guidance provides a compilation in a single document of the pertinent portions of these applicable references that are needed in addressing spent fuel pool modifications. No additional regulatory requirements are imposed or implied by this document.

Based on a review of license applications to date requesting authorization to increase spent fuel storage capacity, the staff has had to request additional information that could have been included in an adequately documented initial submittal. If in the future you find it necessary to apply for authorization to modify onsite spent fuel storage capacity, the enclosed guidance provides the necessary information and acceptance criteria utilized by the NRC staff in evaluating these applications. Providing the information needed to evaluate the matters covered by this document would likely avoid the necessity for NRC questions and thus significantly shorten the time required to process a fuel pool modification amendment.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. NRC Guidance
2. Notice

ENCLOSURE NO. 1

OT POSITION FOR REVIEW AND ACCEPTANCE OF SPENT FUEL STORAGE AND HANDLING APPLICATIONS

I. BACKGROUND

Prior to 1975, low density spent fuel storage racks were designed with a large pitch, to prevent fuel pool criticality even if the pool contained the highest enrichment uranium in the light water reactor fuel assemblies. Due to an increased demand on storage space for spent fuel assemblies, the more recent approach is to use high density storage racks and to better utilize available space. In the case of operating plants the new rack system interfaces with the old fuel pool structure. A proposal for installation of high density storage racks may involve a plant in the licensing stage or an operating plant. The requirements of this position do not apply to spent fuel storage and handling facilities away from the nuclear reactor complex.

On September 16, 1975, the Commission announced (40 F. R. 42801) its intent to prepare a generic environmental impact statement on handling and storage of spent fuel from light water power reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement.

The Commission directed that in the consideration of any such proposed licensing action, an environmental impact statement or environmental impact appraisal shall be prepared in which five specific factors in addition to the normal cost/benefit balance and environmental stresses should be applied, balanced and weighed.

The overall design objectives of a fuel storage facility at the reactor complex are governed by various Regulatory Guides, the Standard Review Plan, and industry standards which are listed in the reference section. Based on the reviews of such applications to date it is obvious that the staff had to request additional information that could be easily included in an adequately documented initial submittal. It is the intent of this document to provide guidance for the type and extent of information needed to perform the review, and to indicate the acceptance criteria where applicable.

II. REVIEW DISCIPLINES

The objective of the staff review is to prepare (1) Safety Evaluation Report, and (2) Environmental Impact Appraisal. The broad staff disciplines involved are nuclear, mechanical, material, structural, and environmental.

Nuclear and thermal-hydraulic aspects of the review include the potential for inadvertant criticality in the normal storage and handling of the spent fuel, and the consequences of credible accidents with respect to criticality and the ability of the heat removal system to maintain sufficient cooling.

Mechanical, material and structural aspects of the review concern the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand the effects of natural phenomena such as earthquakes, tornadoes, flood, effects of external and internal missiles, thermal loading, and also other service loading conditions.

The environmental aspects of the review concern the increased thermal and radiological releases from the facility under normal as well as accident conditions, the occupational radiation exposures, the generation of radioactive waste, the need for expansion, the commitment of material and nonmaterial resources, realistic accidents, alternatives to the proposed action and the cost-benefit balance.

The information related to nuclear and thermal-hydraulic type of analyses is discussed in Section III.

The mechanical, material, and structural related aspects of information are discussed in Section IV.

The information required to complete an environmental impact assessment, including the five factors specified by the Commission, is provided in Section V.

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

1. Neutron Multiplication Factor

To include all credible conditions, the licensee shall calculate the effective neutron multiplication factor, k_{eff}, in the fuel storage pool under the following sets of assumed conditions:

1.1 Normal Storage

a. The racks shall be designed to contain the most reactive fuel authorized to be stored in the facility without any control rods or any noncontained* burnable poison and the fuel shall be assumed to be at the most reactive point in its life.

b. The moderator shall be assumed to be pure water at the temperature within the fuel pool limits which yields the largest reactivity.

c. The array shall be assumed to be infinite in lateral extent or to be surrounded by an infinitely thick water reflector and thick concrete,** as appropriate to the design.

d. Mechanical uncertainties may be treated by assuming "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

e. Credit may be taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption, provided a means of inspection is established (refer to Section 1.5).

1.2 Postulated Accidents

The double contingency principle of ANSI N 16.1-1975 shall be applied. It shall require two unlikely, independent, concurrent events to produce a criticality accident.

Realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies. The

*"Noncontained" burnable poison is that which is not an integral part of the fuel assembly.

**It should be noted that under certain conditions concrete may be a more effective reflector than water.

postulated accidents shall include: (1) dropping of a fuel element on top of the racks and any other achievable abnormal location of a fuel assembly in the pool; (2) a dropping or tipping of the fuel cask or other heavy objects into the fuel pool; (3) effect of tornado or earthquake on the deformation and relative position of the fuel racks; and (4) loss of all cooling systems or flow under the accident conditions, unless the cooling system is single failure proof.

1.3 Calculation Methods

The calculation method and cross-section values shall be verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. Sufficiently diverse configurations shall be calculated to render improbable the "cancellation of error" in the calculations. So far as practicable the ability to correctly account for heterogeneities (e.g., thin slabs of absorber between storage locations) shall be demonstrated.

A calculational bias, including the effect of wide spacing between assemblies shall be determined from the comparison between calculation and experiment. A calculation uncertainity shall be determined such that the true multiplication factor will be less than the calculated value with a 95 percent probability at a 95 percent confidence level. The total uncertainity factor on k_{eff} shall be obtained by a statistical combination of the calculational and mechanical uncertainties. The k_{eff} value for the racks shall be obtained by summing the calculated value, the calculational bias, and the total uncertainty.

1.4 Rack Modification

For modification to existing racks in operating reactors, the following information should be provided in order to expedite the review:

(a) The overall size of the fuel assembly which is to be stored in the racks and the fraction of the total cell area which represents the overall fuel assembly in the model of the nominal storage lattice cell;

(b) For H_2O + stainless steel flux trap lattices; the nominal thickness and type of stainless steel used in the storage racks and the thermal (.025 ev) macroscopic neutron absorption cross section that is used in the calculation method for this stainless steel;

(c) Also, for the H_2O + stainless steel flux trap lattices, the change of the calculated neutron multiplication factor of

infinitely long fuel assemblies in infinitely large arrays in the storage rack (i.e., the k_∞ of the nominal fuel storage lattice cell and the changed k_∞) for:

(1) A change in fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly where it is assumed that this change is made by increasing the enrichment of the U^{235}; and,

(2) A change in the thickness of stainless steel in the storage racks assuming that a decrease in stainless steel thickness is taken up by an increase in water thickness and vice versa;

(d) For lattices which use boron or other strong neutron absorbers provide:

(1) The effective areal density of the boron-ten atoms (i.e., B^{10} atoms/cm^2 or the equivalent number of boron-ten atoms for other neutron absorbers) between fuel assemblies.

(2) Similar to Item C, above, provide the sensitivity of the storage lattice cell k_∞ to:

(a) The fuel loading in grams of U^{235}, or equivalent, per axial centimeter of fuel assembly,

(b) The storage lattice pitch; and,

(c) The areal density of the boron-ten atoms between fuel assemblies.

1.5 Acceptance Criteria for Criticality

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

(1) For those facilities which employ a strong neutron absorbing material to reduce the neutron multiplication factor for the storage pool, the licensee shall provide the description of onsite tests which will be performed to confirm the presence and retention of the strong absorber in the racks. The results of an initial, onsite verification test shall show within 95 percent confidence limits that there is a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95. In addition, coupon or other type of surveillance testing shall be performed on a statistically acceptable sample size on a

periodic basis throughout the life of the racks to verify the continued presence of a sufficient amount of neutron absorber in the racks to maintain the neutron multiplication factor at or below 0.95.

(2) Decay Heat Calculations for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system shall be made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 75/087).

(3) Thermal-Hydraulic Analyses for Spent Fuel Cooling

Conservative methods should be used to calculate the maximum fuel temperature and the increase in temperature of the water in the pool. The maximum void fraction in the fuel assembly and between fuel assemblies should also be calculated.

Ordinarily, in order not to exceed the design heat load for the spent fuel cooling system it will be necessary to do a certain amount of cooling in the reactor vessel after reactor shutdown prior to moving fuel assemblies into the spent fuel pool. The bases for the analyses should include the established cooling times for both the usual refueling case and the full core off load case.

A potential for a large increase in the reactivity in an H_2O flux trap storage lattice exists if, somehow, the water is kept out or forced out of the space between the fuel assemblies, conceivably by trapped air or steam. For this reason, it is necessary to show that the design of the storage rack is such that this will not occur and that these spaces will always have water in them. Also, in some cases, direct gamma heating of the fuel storage cell walls and of the intercell water may be significant. It is necessary to consider direct gamma heating of the fuel storage cell walls and of the intercell water to show that boiling will not occur in the water channels between the fuel assemblies. Under postulated accident conditions where all non-Category I spent fuel pool cooling systems become inoperative, it is necessary to show that there is an alternate method for cooling the spent pool water. When this alternative method requires the installation of alternate components or significant physical alteration of the cooling system, the detailed steps shall be described, along with the time required for each. Also, the average amount of water in the fuel pool and the expected heat up rate of this water assuming loss of all cooling systems shall be specified.

(4) Potential Fuel and Rack Handling Accidents

The method for moving the racks to and from and into and out of the fuel pool, should be described. Also, for plants where the spent fuel pool modification requires different fuel handling procedures than that described in the Final Safety Analysis Report, the differences should be discussed. If potential fuel and rack handling accidents occur, the neutron multiplication factor in the fuel pool shall not exceed 0.95. These postulated accidents shall not be the cause of the loss of cooling for either the spent fuel or the reactor.

(5) Technical Specifications

To insure against criticality, the following technical specifications are needed on fuel storage in high density racks:

1. The neutron multiplication factor in the fuel pool shall be less than or equal to 0.95 at all times.

2. The fuel loading (i.e., grams of uranium-235, or equivalent, per axial centimeter of assembly) in fuel assemblies that are to be loaded into the high density racks should be limited. The number of grams of uranium-235, or equivalent, put in the plant's technical specifications shall preclude criticality in the fuel pool.

Excessive pool water temperatures may lead to excessive loss of water due to evaporation and/or cause fogging. Analyses of thermal load should consider loss of all pool cooling systems. To avoid exceeding the specified spent fuel pool temperatures, consideration shall be given to incorporating a technical specification limit on the pool water temperature that would resolve the concerns described above. For limiting values of pool water temperatures refer to ANSI-N210-1976 entitled, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," except that the requirements of the Section 9.1.3.III.1.d of the Standard Review Plan is applicable for the maximum heat load with normal cooling systems in operation.

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

(1) Description of the Spent Fuel Pool and Racks

Descriptive information including plans and sections showing the spent fuel pool in relation to other plant structures shall be provided in order to define the primary structural aspects and elements relied upon to perform the safety-related functions of the pool and the racks. The main safety function of the spent fuel pool and the racks is to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings, such as earthquake, and impact due to spent fuel cask drop, drop of a spent fuel assembly, or drop of any other heavy object during routine spent fuel handling.

The major structural elements reviewed and the extent of the descriptive information required are indicated below.

(a) Support of the Spent Fuel Racks: The general arrangements and principal features of the horizontal and the vertical supports to the spent fuel racks should be provided indicating the methods of transferring the loads on the racks to the fuel pool wall and the foundation slab. All gaps (clearance or expansion allowance) and sliding contacts should be indicated. The extent of interfacing between the new rack system and the old fuel pool walls and base slab should be discussed, i.e., interface loads, response spectra, etc.

If connections of the racks are made to the base and to the side walls of the pool such that the pool liner may be perforated, the provisions for avoiding leakage of radioactive water of the pool should be indicated.

(b) Fuel Handling: Postulation of a drop accident, and quantification of the drop parameters are reviewed under the environmental discipline. Postulated drop accidents must include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. Integrity of the racks and the fuel pool due to a postulated fuel handling accident is reviewed under the mechanical, material, and structural disciplines. Sketches and sufficient details of the fuel handling system should be provided to facilitate this review.

(2) Applicable Codes, Standards and Specifications

Construction materials should conform to Section III, Subsection NF of the ASME* Code. All Materials should be selected to be compatible with the fuel pool environment to minimize corrosion and galvanic effects.

Design, fabrication, and installation of spent fuel racks of stainless steel material may be performed based upon the AISC** specification or Subsection NF requirements of Section III of the ASME B&PV Code for Class 3 component supports. Once a code is chosen its provisions must be followed in entirety. When the AISC specification procedures are adopted, the yield stress values for stainless steel base metal may be obtained from the Section III of the ASME B&PV Code, and the design stresses defined in the AISC specifications as percentages of the yield stress may be used. Permissible stresses for stainless steel welds used in accordance with the AISC Code may be obtained from Table NF-3292.1-1 of ASME Section III Code.

Other materials, design procedures, and fabrication techniques will be reviewed on a case by case basis.

(3) Seismic and Impact Loads

For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61 respectively. For plants where dynamic data are available, e.g., ground response spectra for a fuel pool supported by the ground, floor response spectra for fuel pools supported on soil where soil-structure interaction was considered in the pool design or a floor response spectra for a fuel pool supported by the reactor building, the design and analysis of the new rack system may be performed by using either the existing input parameters including the old damping values or new parameters in accordance with Regulatory Guide 1.60 and 1.61. The use of existing input with new damping values in Regulatory Guide 1.61 is not acceptable.

Seismic excitation along three orthogonal directions should be imposed simultaneously for the design of the new rack system.

*American Society of Mechanical Engineers Boiler and Pressure Vessel Codes, Latest Edition.

**American Institute of Steel Construction, Latest Edition.

The peak response from each direction should be combined by square root of the sum of the squares. If response spectra are available for a vertical and horizontal directions only, the same horizontal response spectra may be applied along the other horizontal direction.

The effect of submergence of the rack system on the damping and the mass of the fuel racks has been under study by the NRC. Submergence in water may introduce damping from two sources, (a) viscous drag, and (b) radiation of energy away from the submerged body in those cases where the confining boundaries are far enough away to prevent reflection of waves at the boundaries. Viscous damping is generally negligible. Based upon the findings of this current study for a typical high density rack configuration, wave reflections occur at the boundaries so that no additional damping should be taken into account.

A report on the NRC study is to be published shortly under the title "Effective Mass and Damping of Submerged Structures (UCRL-52342)," by R. G. Dong. The recommendations provided in this report on the added mass effect provide an acceptable basis for the staff review. Increased damping due to submergence in water is not acceptable without applicable test data and/or detailed analytical results.

Due to gaps between fuel assemblies and the walls of the guide tubes, additional loads will be generated by the impact of fuel assemblies during a postulated seismic excitation. Additional loads due to this impact effect may be determined by estimating the kinetic energy of the fuel assembly. The maximum velocity of the fuel assembly may be estimated to be the spectral velocity associated with the natural frequency of the submerged fuel assembly. Loads thus generated should be considered for local as well as overall effects on the walls of the rack and the supporting framework. It should be demonstrated that the consequent loads on the fuel assembly do not lead to a damage of the fuel.

Loads generated from other postulated impact events may be acceptable, if the following parameters are described in the report: the total mass of the impacting missile, the maximum velocity at the time of impact, and the ductility ratio of the target material utilized to absorb the kinetic energy.

(4) Loads and Load Combinations:

Any change in the temperature distribution due to the proposed modification should be identified. Information pertaining to the applicable design loads and various combinations thereof should be provided indicating the thermal load due to the effect of the maximum temperature distribution through the pool walls and base

slab. Temperature gradient across the rack structure due to differential heating effect between a full and an empty cell should be indicated and incorporated in the design of the rack structure. Maximum uplift forces available from the crane should be indicated including the consideration of these forces in the design of the racks and the analysis of the existing pool floor, if applicable.

The specific loads and load combinations are acceptable if they are in conformity with the applicable portions of Section 3.8.4-II.3 of the Standard Review Plan.

(5) Design and Analysis Procedures

Details of the mathematical model including a description of how the important parameters are obtained should be provided including the following: the methods used to incorporate any gaps between the support systems and gaps between the fuel bundles and the guide tubes; the methods used to lump the masses of the fuel bundles and the guide tubes; the methods used to account for the effect of sloshing water on the pool walls; and, the effect of submergence on the mass, the mass distribution and the effective damping of the fuel bundle and the fuel racks.

The design and analysis procedures in accordance with Section 3.8.4-II.4 of the Standard Review Plan are acceptable. The effect on gaps, sloshing water, and increase of effective mass and damping due to submergence in water should be quantified.

When pool walls are utilized to provide lateral restraint at higher elevations, a determination of the flexibility of the pool walls and the capability of the walls to sustain such loads should be provided. If the pool walls are flexible (having a fundamental frequency less than 33 Hertz), the floor response spectra corresponding to the lateral restraint point at the higher elevation are likely to be greater than those at the base of the pool. In such a case using the response spectrum approach, two separate analyses should be performed as indicated below:

(a) A spectrum analysis of the rack system using response spectra corresponding to the highest support elevation provided that there is not significant peak frequency shift between the response spectra at the lower and higher elevations; and,

(b) A static analysis of the rack system by subjecting it to the maximum relative support displacement.

The resulting stresses from the two analyses above should be combined by the absolute sum method.

In order to determine the flexibility of the pool wall it is acceptable for the licensee to use equivalent mass and stiffness properties obtained from calculations similar to those described "Introduction to Structural Dynamics" by J. M. Biggs published by McGraw Hill Book Company. Should the fundamental frequency of the pool wall model be higher than or equal to 33 Hertz, it may be assumed that the response of the pool wall and the corresponding lateral support to the new rack system are identical to those of the base slab, for which appropriate floor response spectra or ground response spectra may already exist.

(6) Structural Acceptance Criteria

When AISC Code procedures are adopted, the structural acceptance criteria are those given in Section 3.8.4.II.5 of the Standard Review Plan for steel and concrete structures. For stainless steel the acceptance criteria expressed as a percentage of yield stress should satisfy Section 3.8.4.II.5 of the Standard Review Plan. When subsection NF, Section III, of the ASME B&PV Code is used for the racks, the structural acceptance criteria are those given in the Table below.

For impact loading the ductility ratios utilized to absorb kinetic energy in the tensile, flexural, compressive, and shearing modes should be quantified. When considering the effects of seismic loads, factors of safety against gross sliding and overturning of racks and rack modules under all probable service conditions shall be in accordance with the Section 3.8.5.II-5 of the Standard Review Plan. This position on factors of safety against sliding and tilting need not be met provided any one of the following conditions is met:

(a) it can be shown by detailed nonlinear dynamic analyses that the amplitudes of sliding motion are minimal, and impact between adjacent rack modules or between a rack module and the pool walls is prevented provided that the factors of safety against tilting are within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

(b) it can be shown that any sliding and tilting motion will be contained within suitable geometric constraints such as thermal clearances, and that any impact due to the clearances is incorporated.

(7) Materials, Quality Control, and Special Construction Techniques:

The materials, quality control procedures, and any special construction techniques should be described. The sequence of installation of the new fuel racks, and a description of the precautions to be taken to prevent damage to the stored fuel during

TABLE

Load Combination	
Elastic Analysis	**Acceptance Limit**
D + L	Normal limits of NF 3231.1a
D + L + E	Normal limits of NF 3231.1a
D + L + To	1.5 times normal limits or the lesser of 2 Sy and Su
D + L + To + E	1.5 times normal limits or the leser of 2 Sy and Su
D + L + Ta + E	1.6 times normal limits or the lesser of 2 Sy or Su
D + L + Ta + E[1]	Faulted condition limits of NF 3231.1c
Limit Analysis	
1.7 (D + L)	Limits of XVII-4000 of Appendix XVII of ASME Code Section III
1.7 (D + L + E)	
1.3 (D + L + To)	
1.3 (D + L + E + To)	
1.1 (D + L + Ta + E)	

Notes:

1. The abbreviations in the table above are those used in Section 3.8.4 of the Standard Review Plan where each term is defined except for Ta which is defined as the highest temperature associated with the postulated abnormal design conditions.

2. Deformation limits specified by the Design Specification limits shall be satisfied, and such deformation limits should preclude damage to the fuel assemblies.

3. The provisions of NF 3231.1 shall be ammended by the requirements of the paragraphs c.2, 3, and 4 of the Regulatory Guide 1.124 entitled "Design Limits and Load Combinations for Class 1 Linear-Type Component Supports."

the construction phase should be provided. Methods for structural qualification of special poison materials utilized to absorb neutron radiation should be described. The material for the fuel rack is reviewed for compatibility inside the fuel pool environment. The quality of the fuel pool water in terms of the pH value and the available chlorides, fluorides, boron, heavy metals should be indicated so that the long-term integrity of the rack structure, fuel assembly, and the pool liner can be evaluated.

Acceptance criteria for special materials such as poison materials should be based upon the results of the qualification program supported by test data and/or analytical procedures.

If connections between the rack and the pool liner are made by welding, the welder as well as the welding procedure for the welding assembly shall be qualified in accordance with the applicable code.

If precipitation hardened stainless steel material is used for the construction of the spent fuel pool racks, hardness testing should be performed on each rack component of the subject material to verify that each part is heat treated properly. In addition, the surface film resulting from the heat treatment should be removed from each piece to assure adequate corrosion resistance.

(8) Testing and Inservice Surveillance

Methods for verification of long-term material stability and mechanical integrity of special poison material utilized for neutron absorption should include actual tests.

Inservice surveillance requirements for the fuel racks and the poison material, if applicable, are dependent on specific design features. These features will be reviewed on a case by case basis to determine the type and the extent of inservice surveillance necessary to assure long-term safety and integrity of the pool and the fuel rack system.

V. COST/BENEFIT ASSESSMENT

1. Following is a list of information needed for the environmental Cost/Benefit Assessment:

1.1 What are the specific needs that require increased storage capacity in the spent fuel pool (SFP)? Include in the response:

(a) status of contractual arrangements, if any, with fuel-storage or fuel-reprocessing facilities,

(b) proposed refueling schedule, including the expected number of fuel assemblies that will be transferred into the SFP at each refueling until the total existing capacity is reached,

(c) number of spent fuel assemblies presently stored in the SFP,

(d) control rod assemblies or other components stored in the SFP, and

(e) the additional time period that spent fuel assemblies would be stored onsite as a result of the proposed expansion, and

(f) the estimated date that the SFP will be filled with the proposed increase in storage capacity.

1.2 Discuss the total construction associated with the proposed modification, including engineering, capital costs (direct and indirect) and allowances for funds used during construction.

1.3 Discuss the alternative to increasing the storage capacity of the SFP. The alternatives considered should include:

(a) shipment to a fuel reprocessing facility (if available),

(b) shipment to an independent spent fuel storage facility,

(c) shipment to another reactor site,

(d) shutting down the reactor.

The discussion of options (a), (b) and (c) should include a cost comparison in terms of dollars per KgU stored or cost per assembly. The discussion of (d) should include the cost for providing replacement power either from within or outside the licensee's generating system.

1.4 Discuss whether the commitment of material resources (e.g., stainless steel, boral, B_4C, etc.) would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. Describe the material resources that would be consumed by the proposed modification.

1.5 Discuss the additional heat load and the anticipated maximum temperature of water in the SFP which would result from the proposed expansion, the resulting increase in evaporation rates, the additional heat load on component and/or plant cooling water systems and whether there will be any significant increase in the amount of heat released to the environment.

V.2. RADIOLOGICAL EVALUATION

2. Following is a list of information needed for radiological evaluation:

2.1 The present annual quantity of solid radioactive wastes generated by the SFP purification system. Discuss the expected increase in solid wastes which will result from the expansion of the capacity of the SFP.

2.2. Data regarding krypton-85 measured from the fuel building ventilation system by year for the last two years. If data are not available from the fuel building ventilation system, provide this data for the ventilation release which includes this system.

2.3 The increases in the doses to personnel from radionuclide concentrations in the SFP due to the expansion of the capacity of the SFP, including the following:

(a) Provide a table showing the most recent gamma isotopic analysis of SFP water identifying the principal radionuclides and their respective concentrations.

(b) The models used to determine the external dose equivalent rate from these radionuclides. Consider the dose equivalent rate at some distance above the center and edge of the pool respectively. (Use relevant experience if necessary).

(c) A table of recent analysis performed to determine the principal airborne radionuclides and their respective concentrations in the SFP area.

(d) The model and assumptions used to determine the increase, if any, in dose rate from the radionuclides identified in (c) above in the SFP area and at the site boundary.

(e) An estimate of the increase in the annual man-rem burden from more frequent changing of the demineralizer resin and filter media.

(f) The buildup of crud (e.g., ^{58}Co, ^{60}Co) along the sides of the pool and the removal methods that will be used to reduce radiation levels at the pool edge to as low as reasonably achievable.

(g) The expected total man-rem to be received by personnel occupying the fuel pool area based on all operations in that area including the doses resulting from (e) and (f) above.

A discussion of the radiation protection program as it affects (a) through (g) should be provided.

2.4 Indicate the weight of the present spent fuel racks that will be removed from the SFP due to the modification and discuss what will be done with these racks.

V.3 ACCIDENT EVALUATION

3.1 The accident review shall consider:

(a) cask drop/tip analysis, and

(b) evaluation of the overhead handling system with respect to Regulatory Guide 1.104.

3.2 If the accident aspects of review do not establish acceptability with respect to either (a) or (b) above, then technical specifications may be required that prohibit cask movement in the spent fuel building.

3.3 If the accident review does not establish acceptability with respect to (b) above, then technical specifications may be required that:

(1) define cask transfer path including control of

(a) cask height during transfer, and

(b) cask lateral position during transfer

(2) indicate the minimum age of fuel in pool sections during movement of heavy loads near the pool. In special cases evaluation of consequences-limiting engineered safety features such as isolation systems and filter systems may be required.

3.4 If the cask drop/tip analysis as in 3.1(a) above is promised for future submittal, the staff evaluation will include a conclusion on the feasibility of a specification of minimum age of fuel based on previous evaluations.

3.5 The maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. A technical specification will be required to this effect.

3.6 Conclusions that determination of previous Safety Evaluation Reports and Final Environmental Statements have not changed significantly or impacts are not significant are made so that a negative declaration with an Environmental Impact Appraisal (rather than a Draft and Final Environmental Statement) can be issued. This will involve checking realistic as well as conservative accident analyses.

VI. REFERENCES

1. Regulatory Guides

   1.13 - Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations

   1.29 - Seismic Design Classification

   1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

   1.61 - Damping Values for Seismic Design of Nuclear Power Plants

   1.76 - Design Basis Tornado for Nuclear Power Plants

   1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

   1.104 - Overhead Crane Handling Systems for Nuclear Power Plants

   1.124 - Design Limits and Loading Combinations for Class 1 Linear-Type Components Supports

2. Standard Review Plan

   3.7 - Seismic Design

   3.8.4 - Other Category I Structures

   9.1 - Fuel Storage and Handling

   9.5.1 - Fire Protection System

3. Industry Codes and Standards

   1. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Section III, Division 1

   2. American Institute of Steel Construction Specifications

   3. American National Standards Institute, N210-76

   4. American Society of Civil Engineers, Suggested Specification for Structures of Aluminium Alloys 6061-T6 and 6067-T6

5. The Aluminium Association, Specification for Aluminium Structures

EXHIBIT 3

Draft 1, Regulatory Guide 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis)(December 1981)

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard that details minimum design requirements for

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAR 5 1982

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

FOR COMMENT

U.S. NUCLEAR REGULATORY COMMISSION

spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," by the American National Standards Institute on April 12, 1976.

Primary facility design objectives are:

a. To prevent loss of water from the fuel pool that would uncover fuel,

b. To protect the spent fuel from mechanical damage, and

c. To provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. LOSS OF WATER FROM STORAGE POOL

Unless protective measures are taken to prevent the loss of water from a fuel storage pool, the spent fuel could overheat and cause damage to fuel cladding integrity, which could result in the release of radioactive materials to the environment. Equipment failures in systems connected to the pool could also result in the loss of pool water. A permanent coolant makeup system designed with suitable redundancy or backup would prevent the fuel from being uncovered should pool leaks occur. Further, early detection of pool leakage and fuel damage can be made using pool-water-level monitors and pool radiation monitors that alarm locally and also at a continuously manned location to ensure timely operation of building filtration systems. Natural events such as earthquakes or high winds can damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity will alleviate these concerns.

2. MECHANICAL DAMAGE TO FUEL

The release of radioactive material from fuel may occur as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or objects onto fuel elements during the refueling process and at other times.

Plant arrangements consider low-probability accidents such as the dropping of heavy loads (e.g., a 100-ton fuel cask) where such loads are positioned or moved in or over the spent fuel pool. It is desirable that cranes capable of carrying heavy loads be prevented from moving into the vicinity of the stored fuel.

Missiles generated by high winds also are a potential cause of mechanical damage to fuel. This concern can be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. LIMITING POTENTIAL OFFSITE EXPOSURES

Mechanical damage to the fuel might cause significant offsite doses unless dose reduction features are provided. Dose reduction designs such as negative pressure in the fuel handling building during movement of spent fuel would prevent exfiltration and ensure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment. Even if measures not described are used to maintain the desired negative pressure, small leaks from the building may still occur as a result of structural failure or other unforeseen events.

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protection requirements are acceptable for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the filtration-ventilation system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

Power Plants," provide guidelines to limit potential offsite exposures through the filtration-ventilation system of the pool building.

Occupational radiation exposure is kept as low as is reasonably achievable (ALARA) in all activities involving personnel, and efforts toward maintaining exposures ALARA are considered in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

The requirements in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations,"* are generally acceptable to the NRC staff as a means for complying with the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 as related to light-water reactors (LWRs), subject to the following clarifications and modifications:

1. In lieu of the example inventory in Section 4.2.4.3(1), the example inventory should be that inventory of radioactive materials that are predicted to leak under the postulated maximum damage conditions resulting from the dropping of a single spent fuel assembly onto a fully loaded spent fuel pool storage rack. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

3. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to prevent tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to prevent missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado-generated missiles or from seismic damage to ensure that nothing bypasses the ESF-grade filtration system in the containment building.

4. In addition to meeting the requirements of Section 5.1.5.1, provisions should be made to ensure that nonfuel components in fuel pools are handled below the minimum water shielding depth. A system should be provided that, either through the design of the system or through administrative procedures, would prohibit unknowing retrieval of these components.

5. In addition to meeting the requirements of Section 5.1.12.10, the maximum potential kinetic energy capable of being developed by any object handled above stored spent fuel, if dropped, should not exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface should be provided between the cask venting system and the building ventilation system to minimize personnel exposure to the "vent-gas" generated from filling a dry loaded cask with water.

7. In addition to meeting the requirements of Section 5.3.3, radioactivity released during a Condition IV fuel handling accident should be either contained or removed by filtration so that the dose to an individual is less than the guidelines of 10 CFR Part 100. The calculated offsite dose to an individual from such an event should be well within the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to ensure that released activity does not bypass the

filtration system, the ESF fuel storage building ventilation should provide and maintain a negative pressure of at least 3.2 mm (0.125 in.) water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed so that travel directly over the spent fuel storage pool or safety-related equipment is not possible. This should be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I, and safety-related structures and equipment should be subjected to quality assurance programs that meet the applicable provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Further, these programs should obtain guidance from Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," endorsing ANSI N45.2, and from the applicable provisions of the ANSI N45.2-series standards endorsed by the following regulatory guides:

1.30 (Safety Guide 30) "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (N45.2.4).

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (N45.2.2).

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (N45.2.6).

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants" (N45.2.11).

1.74 "Quality Assurance Terms and Definitions" (N45.2.10).

1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (N45.2.9).

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (N45.2.5).

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (N45.2.8).

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (N45.2.13).

10. The spent fuel pool water temperatures stated in Section 6.6.1(2) exceed the limits recommended by the NRC staff. For the maximum heat load during Condition I occurrences with normal cooling systems in operation and assuming a single active failure, the pool water temperature should be kept at or below 60°C (140°F). Under abnormal maximum heat load conditions (full core unload) and also for Condition IV occurrences, the pool water temperature should be kept below boiling.

11. A nuclear criticality safety analysis should be performed in accordance with Appendix A to this guide for each system that involves the handling, transfer, or storage of spent fuel assemblies at LWR spent fuel storage facilities.

12. The spent fuel storage facility should be equipped with both electrical interlocks and mechanical stops to keep casks from being transported over the spent fuel pool.

13. Sections 6.4 and 9 of ANS-57.2 list those codes and standards referenced in ANS-57.2. Although this regulatory guide endorses with clarifications and modifications ANS-57.2, a blanket endorsement of those referenced codes and

standards is not intended. (Other regulatory guides may contain some such endorsements.)

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants regarding the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 30, 1982.

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

1.1 A nuclear criticality safety analysis should be performed for each system that involves the handling, transfer, or storage of spent fuel assemblies at light-water reactor (LWR) spent fuel storage facilities.

1.2 The nuclear criticality safety analysis should demonstrate that each LWR spent fuel storage facility system is subcritical (k_{eff} not to exceed 0.95).

1.3 The nuclear criticality safety analysis should include consideration of all credible normal and abnormal operating occurrences, including:

a. Accidental tipping or falling of a spent fuel assembly,
b. Accidental tipping or falling of a storage rack during transfer,
c. Misplacement of a spent fuel assembly,
d. Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system,
e. Fuel drop accidents,
f. Stuck fuel assembly/crane uplifting forces,
g. Horizontal motion of fuel before complete removal from rack,
h. Placing a fuel assembly along the outside of rack, and
i. Objects that may fall onto the stored spent fuel assemblies.

1.4 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.

1.5 The nuclear criticality safety analysis should explicitly identify spent fuel assembly characteristics upon which subcriticality in the LWR spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis should explicitly identify design limits upon which subcriticality depends that require physical verification at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis should explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

2. CALCULATION METHODS AND CODES

Methods used to calculate subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI N16.9-1975.

3. METHOD TO ESTABLISH SUBCRITICALITY

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks, k_s, under normal and credible abnormal conditions should be equal to or less than an established maximum allowable multiplication factor, k_a; i.e.,

$$k_s \leq k_a$$

The factor, k_s, should be evaluated from the expression:

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc}$$

where

k_{sn} = the computed effective multiplication factor; k_{sn} is calculated by the same methods used for benchmark experiments for design storage parameters when the racks are loaded with the most reactive fuel to be stored,

Δk_{sb} = the bias in the calculation procedure as obtained from the comparisons with experiments and including any extrapolation to storage pool conditions,

Δk_u = the uncertainty in the benchmark experiments, and

Δk_{sc} = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, Δk_{sc}, include:

a. Statistical uncertainty in the calculated result if a Monte Carlo calculation is used,

b. Uncertainty resulting from comparison with calculational and experimental results,

c. Uncertainty in the extrapolation from experiment to storage rack conditions, and

d. Uncertainties introduced by the considerations enumerated in paragraphs 4.3 and 4.4 below.

3.3 The various uncertainties may be combined statistically if they are independent. Correlated uncertainties should be combined additively.

3.4 All uncertainty values should be at the 95 percent probability level with a 95 percent confidence value.

3.5 For spent fuel storage pool, the value of k_a should be no greater than 0.95.

4. STORAGE RACK ANALYSIS ASSUMPTIONS

4.1 The spent fuel storage rack module design should be based on one of the following assumptions for the fuel:

a. The most reactive fuel assembly to be stored at the most reactive point in the assembly life, or

b. The most reactive fuel assembly to be stored based on a minimum confirmed burnup (see Section 6 of this appendix).

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly includes consideration of the following parameters:

a. Maximum fissile fuel loading,
b. Fuel rod diameter,
c. Fuel rod cladding material and thickness,
d. Fuel pellet density,
e. Fuel rod pitch and total number of fuel rods within assembly,
f. Absence of fuel rods in certain locations, and
g. Burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design should be the arrangement that results in the highest value of k_s considering:

a. Spacing between assemblies,
b. Moderation between assemblies, and
c. Fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of k_s shall include consideration of the following:

a. Eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,
b. Dimensional tolerances,
c. Construction materials,
d. Fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e. Presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f. Presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Fuel burnup determination should be made for fuel stored in racks where credit is taken for burnup. The following methods are acceptable:

a. A minimum allowable fuel assembly reactivity should be established, and a reactivity measurement should be performed to ensure that each assembly meets this criterion; or

b. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and a measurement should be performed to ensure that each fuel assembly meets the established criterion; or

c. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and an analysis of each fuel assembly's exposure history should be performed to determine its burnup. The analyses should be performed under strict administrative control using approved written procedures. These procedures should provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in paragraph 1.4 above.

The uncertainties in determining fuel assembly storage acceptance criteria should be considered in establishing storage rack reactivity, and auditable records should be kept of the method used to determine the fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration should be given to the axial distribution of burnup in the fuel assembly, and a limit should be set on the length of the fuel assembly that is permitted to have a lower average burnup than the fuel assembly average.

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a. The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b. Fixed neutron absorbers should be an integral, nonremovable part of the storage rack.

c. When a fixed neutron absorber is used as the primary nuclear criticality safety control, there should be provision to:

(1) Initially confirm absorber presence in the storage rack, and

(2) Periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water should not normally be used in the evaluation of k_s. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

6.1 Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, ^{235}U depletion, amount of burnable poison, plutonium buildup and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildup are not necessarily the same.

6.2 Consideration should be given to the practical implementation of the spent fuel screening process. Factors to be considered in choosing the screening method should include:

a. Accuracy of the method used to determine storage rack reactivity;
b. Reproducibility of the result, i.e., what is the uncertainty in the result?
c. Simplicity of the procedure; i.e., how much disturbance to other operations is involved?
d. Accountability, i.e., ease and completeness of recordkeeping; and
e. Auditability.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested that this guide be updated.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit or operating license application. Therefore, there should be a minimum number of cases where the applicant and the staff radically disagree on the design criteria.

1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also, recommendations made by Task A-36, which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," would be included.

3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

Authority for this regulatory guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, through the Commission's regulations, in particular, General Design Criterion 61 of Appendix A to 10 CFR Part 50.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

Regulatory Guide 1.13 should be updated.

EXHIBIT 4

Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants (August 19, 1998)

August 19, 1998

MEMORANDUM TO: Timothy Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Laurence Kopp, Sr. Reactor Engineer /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:
As stated

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PDR ORG NRRA
PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref. 4), which states:

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

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper. In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref 5)

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs) The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs

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The determination of the effective multiplication factor, k_{eff}, for the new or spent fuel storage racks should consider and clearly identify the following:

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

   3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

   4. locations in the fuel assembly lattice that are empty or contain nonfuel material

   5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly

   6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used (k_∞), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the k_∞ limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_∞ and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide (B_4C) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked.

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments. This qualifies both the ability of the analyst and the computer environment. The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref. 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays. A comparison with methods of analysis of similar sophistication (e g . transport theory) may be used to augment or extend the range of applicable critical experiment data

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref 8)

The maximum k_{eff} shall be evaluated from the following expression:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(uncert) + \delta k(burnup),$$

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(burnup)	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of k_{eff} should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff} or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

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

4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core. However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k_{eff} shall be no greater than 0 95, including

mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting k_{eff} is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

1. For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:

a With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than or equal to 0 95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

   b. With the spent fuel storage racks loaded with fuel of the maximum permissible activity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

3. The reference criticality safety analysis should also include, as a minimum, the following:

   a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.

   b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.

   c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1.0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations: i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following

---

[1] [*] is the boron concentration required to maintain the 0.95k_{eff} limit without consideration of accidents

a. the effect of eccentric positioning of fuel assemblies within the storage cells

b. the reactivity consequence of including the flow channel in BWR fuel assemblies

5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.

a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.

b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).

c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.

d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption.

e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution.

B. Additional Considerations

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.

2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered. Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions.

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.

4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.

5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the $0.95k_{eff}$ design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.

7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a k_{eff} less than 0.95.

6. REFERENCES

1. NRC, "Standard Review Plan " NUREG-0800, Rev.2, Section 9.1 1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan " NUREG-0800, Rev. 2, Section 9 1.2, "Spent Fuel Storage," July 1981.

3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

5. Westingnouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.

6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

7. Babcock & Wilcox Company, "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, 1981.

8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.

9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.

EXHIBIT 5

Reactor Engineering Logs, Millstone Unit 3, RFO 6

LABORATORY NO

Billy G 500

LABORATORY RECORDKEEPING PROCEDURES

General Instructions

Laboratory Notebooks are the means for keeping a permanent record of the details of an individual's day-to-da and development areas. They provide 1) a basic reference which the individual and others can refer to at a later d with respect to the materials recorded, such as conception of an invention and the date thereof or date of reductio results; and 3) in some instances, compliance with the provisions of many contractual arrangements.

Instructions for Recording Data in the Laboratory Notebook

1. All entries should be made in a legible and orderly manner using permanent ink, preferably black. Make entri that someone else could repeat the experiment.

2. Avoid erasures. If an error is made, cross it out and make the correction immediately thereafter. Cancellatio be initialed, dated and explained (in the margin, if possible, by an appropriate notation).

3. Make sure control page is filled out prior to usage.

4. Complete the table of contents as you make entries in the notebook.

5. State the object and results of each experiment clearly and concisely. Give a complete, factual and self-expla progress of the work and the procedure followed (reference to earlier work done by yourself or another may be a a previous page of the same notebook or the page and number of an earlier notebook). All operating details a reordered, indicating yeilds, conversion, etc., and identifying products. Describe and give quantities of all ma all code numbers and abbreviations.

6. An entry dealing with a conception (invention, idea) must describe the thing conceived (Example: whether it is a a combinatin of compounds, a combination of a compound and a solvent, etc.) as well as the utility for the thi is to be used and the method(s) by which it is to be prepared, including equivalent materials which could be u regard to utility should be stated positively. Work toward practical implementation of an invention should st after disclosure in a notebook.

7. Negative or deprecatory entries should be avoided. (Example: A reaction gives a 2% yield was expected. Do not not work:.) Phraseology which expresses an opinion rather than a positive statement and gratuitous comment This is not an instruction to omit the conclusion in an experiment which had less than the expected results.

8. Each day's work should, whenever possible, be started on a separate page with lines drawn angularly across the pervious page. (This gives legal evidence that additions were not made at a later date.) It is permissible, as reference to an earlier page by reciting "Continued from page . . . ". It is extremely important that each page

9. Each page must be signed an dated by the individual who makes the entry and does the work. In additio witnessed (signed and dated), using the notation "Read and understood", preferably on the same day, but at l The witness should not be connected with the conception, should not have taken part in the experimental work but should understand the technical field of the entry. Record Lab notebook number and project on each pa

10. Where a conception is made by two or more individuals it need only be entered in the notebook of one, but mu by all the conceivers. Ideally, all conceivers should sign on the same day.

11. For microfilming purposes, graphs, charts, analytical data, etc. must be attached to the notebook pages with and should, when unfolded, be kept within the confines of the opened notebook. NO ENTRIES SHOULD BE M BENEATH ATTACHED SHEETS AND NOTHING MUST BE OBSCURED. Leave the heading at the top a for the witnessing signature at the bottom of each page exposed. Inserts should be signed and dated by the pe and witnessed by another to provide the best legal evidence. If materials such as spectra, graphs, etc. are not they must be signed, dated and identified in such a manner as to provide a reference back to the pertinent pa Notebook itself.

12. Report the loss or theft of a research notebook.

13. Following the use of all the pages of each Laboratory Notebook, it must be sent to the Library for record ma table of contents pages are complete. After it has been recorded it will be returned to the individual. An in his possession the Laboratory Notebook immediately preceding the one in which entries are currently being ma Laboratory Notebooks will be kept on the file in the Library or in vault storage. They may then be checked ou of time by authorized individuals.

51' 4" Containment
HP Control Point

Ext.

Pete Rzase Work 860-285-3143
Home 413-357-6148

The dedicated line (party line) # to di
reestablish communications is 6976.

on-site
Director
x 3960

Control Room Backroom 5006
Roger Van Wey 5210
Jerry Mulligan SNET 2899

Frank Kuzmenko 5828
(Communications)

Randy Wilson 0499
(Communications) 6998 backshift

Jack Johanneman 6414

Tom Dagata 5355 ~~6414~~

containment phone in RCA 6284
SFP wall (clean area) 6334

15 PROJECT NAME _____ NO

4/17/99

1830 Crane (New Fuel Handling) maintenance/rep
completed about ~1630. A bushi
transducer was replaced up in c
control boxes on top.

OPS (Mike Kennedy) performed P
OPS Form 3303.61 and Form
SM approved at ~1730.

Secured (W) crew at 1745 with no f

Noted debris in SFP at the middle
Western Racks, but the east side
near the SFP floor just outside
location V35 (approximately).
Kent W. cannot identify it
Appears to be either a parallel
resin or something jutting ou
the racks but leaving a shadow

SM/VS advised and will leave
off tonight. Jim B wants to
to see if light changes the
tomorrow. Will retrieve at lunch

I will be in at 0900, unless called

4/18/99 0650 Called Cortuff (Gene Olson) + verified cra
for use.

0730 Started moving fuel from NFV to

1027 (W) should determine coordinates of

U-1:	294.0	623.9
A-18:	116.3	443.5
V-44:	306.2	178.3
AV-27:	548.5	358.5

These values are read in inches on PL
The numbers will be stored in memory
calibrate matrix.

1855 Completed NFV Row H and stopped f
SFP Bridge Crane Key #59 checked ou

SIGNATURE Kent Whitham DATE
READ AND UNDERSTOOD

PROJECT NAME _____ N

4/22/99 SHEFFROWERS TRAINING
- KONT WIDMAN
- JACK CAMP
- KIMBERLY PORUSKI
- JOHN STODDARD
- KATHY SIMMONS

4/19 (Late Entry) Completed moving all 81 F/G's to SFP
Performed checkoff of control rod drive
tool - needed replacement of 3 on
calibrated tool + determined max
for unlatching to be 182 psig

4/29 Completed safety harness training for RE
RE overview training - Dave McDaniel gave

~~4~~ 5/1 1011 Rx Trip Breakers open - Rx shutdown

5/10 Daystaff today starts 3 shift rotation
the following people - Kathy Bennett
John Stoddard
Jay Eby

The plan for today is to fill out TOR
briefing at 1300 (with Westinghouse also)
monitor Rx level light.

Kont is processing AP20's offload/reload
to be changed to add inmast
hookup and deadout.

Kathy is finishing the RCI's.

1300 Pre job brief conducted w/ (N) refueling crew

1530 Independent verification of offload, shuffle & reload sequence

1700 Tags have been placed on the Fuel bldg roll-up doors

1830 Tagging verified by KMD; SFP level @ 42% on LI-26;
Pre-Op and weekly checks last performed on 5/7/99,
verified closed and labelled IAW SF 248-3

2235 AWOs M396 05319 and M399 05307 given to J Camp - t
by Ops

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17 PROJECT NAME _____ NOTEE

2300 Midnight shift on, attending pre-job brief.

Jeff Camp — Angelo Leone
Mike Baldwin — John Magynik

5/11 Spent the shift completing the Refuel Helper including self-study, walkdowns, briefing.

Rx vessel head lift was completed at abo

5/11/99 Camera equipment installed in sfp ↓ Monitor on clean side. Checked height of camera SF Bridge crane coordinates reprogrammed, dummy digital readout works perfectly.

Cavity filled to normal level. In-mast connection underwater. Westinghouse h sipping - the console was moved to the + SIGMA due to the fact that where it originally installed was in the way of the still investigating whether it can stay on needs to be moved again (Call George information X4365)

1530 Swingshift crew on:
K. Doroski D. Asay K. Connor
S. Stark R. Racicot

For the shift: if Control room is accessible, set up ta
- tour fuel bldg & ctmt with shift
- work on TQRs & FME qualification

Status:
AWOs with (W); SFP Cs 2738 ppm @ 0925 on 5/10; SFP LL-2

Notes: Contact Neil Knutsen re reconciliation of fuel bldg swings & mids

For problems w/ Ericsson phones, contact R. Wilson

1630 Jim Beaupre in ctmt for resolution of issue with plac console.

1730 Jim informed me that sipping console has been reloca bridge (vs. trolley)

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5/11 18 PROJECT NAME _____ NC

1800 Conducted brief of swingshift RE people

1830 Discussed set-up in Control Rm w/ Gene Olson (SH
not complete - Gene asked that someone come
midnight to set up table, computer, etc. He
we set up in the former tagging area where
room. The way I read Tech Specs, we have
Control room proper. Otherwise, seems like a
would have to bean a headset.

1900 Entire shift crew went on walkdown of Fuel bldg
floor. Checked FME log in fuel bldg - looked good

2130 Spoke w/ Aaron E. - unlatch brief schedule now se
will need phone/headset communications betwee
refuel floor. Should be able to set up in Cont
Ray Martin will be on.

2300 Nightcrew on:

J Camp M Baldwin
A Leone J Magyarik

5/12 0000 B EFF LOP test is in progress. Will unlatc
test is done.

0100 Two issues came up. We thought they had
exemption for CRDM unlatch. I said RE
an exemption for our AUO's but we don't
for CRDM unlatch. So I made a copy of
from our AUO for Art Atkinson. He will
NRC. The other issue is the blue tag o
over the blue tag so we will have to get
an TSMA for CRDM unlatch. No problem
told us that once the blue tag is re
TSMA checks are done) we will not be a
switch because ops must operate it. Thi
Its also not true. We can operate it
their procedure covers it.

0500 EFF LOP test is complete with only minor pro
still not allowed (CRDM unlatch) until

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5/12 **19** PROJECT NAME _____ NOTES

0600 Asked Ray Martin about moving our table into the control room. They would not allow EEE (OP. Ray was trying to get us to set in part) in the back room. But OP 3260 Figure showing the boundary of the control it does NOT include the back room. If put some stuff back there, its not a good may be necessary. And since the control is so much smaller now than in previ it will be tough to get the 10 foot wi to fit anywhere. The better solution to get new smaller white boards made.

The camera to input top monitor is in place transfer canal with the video screen and the clean area.

Late entry - sorry Kathy, the solution to room crowding is to use shuttlecock that program is reliable and when fri everyone knows how to use it. like

0700 DAYSHIFT HERE : JAY EGY
Kathy Emmons JOHN CHANIS
MARK NOFINA
CLAUDIA DRAW

0730 ERICSSON PHONE PARTY LINE ESTA
THE FOLLOWING "6" PHONES 6965, 68
6975, 697
MAKE UP THE PARTY LINE
TO ACTIVATE THE PARTY LINE CALL
LET THE PHONES RING AT LEAST
TIMES ~~TO GO~~ BEFORE ANSWERING,
ENSURE THAT ALL STATIONS HERE TH
WITH REPLACING BATTERY, TURN P
REPLACE BATTERY, TURN PHONE ON
YES BUTTON CONNECTION WILL
MADE.

0830 Phones being set up for CRDS unlatch work
(W) Ready to go on station for CRDS wa
so most sipping consoles is in it's final
place - please do NOT move again.

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5/12 20 ~~1830~~ 0700

PROJECT NAME _____ NOTE

0924 Started unlatching driveshafts - on phone Ericsson in Containment (Dale Brodsky

1145 Left ERICSSON Phone re-charge unit ~~left~~ CRD at Point (Cont. 51'4 level) Notified FMEA monit notify the Operator (Dale Brodsky)

1257 CRDS Unlatching Complete 1257 - we unlatched TAW 3303C Step 4.3.33

1300 Conducted TQR Training Session w/ Day Sh Withers, John Chunis, Mark Norcia & Jay Claudio Nieves - All TQR's are complete signed by Mike Jensen

1330 Moved refueling tag boards into Conta set up RE station.

1441 Log entry: Verification of CRDM unlatch completed, preps f

1500 Swingshift on:

K Dorosli, Paul Raucat, Don Asay Sheila Stark,

Evolutions in progress:

- Communications set up in CR: CTMT for Internals lift
- RE station being set up in CR
- Shuffleworks being tested against RWLs

1645 Internals out, SIGMA final checks beginning, expect to eight hours. Also, working out which "land line" in CR - will use Shift Tech's phone (with their p other available.

1810 Aaron called - was not successful setting up party l verifying good batteries, etc and it did not work through, but all others can't hear in due to feedback

1950 Spoke w/ Randy Wilson (home 860-399-7275, MP ext 0 that the problem was a loose headset jack. Found a again.

2000 Comm. est'd, (W) on station in SFP: CTMT SIGMA

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5/12 21 PROJECT NAME _____ NOTEBO

Note: When ready to start offload, page Keith Covin AN
SRO support. (give some lead time ~1 hr)

2030 Dummy in CTMT for SIGMA checks. when finished (
need PEO in SFP to operate gate vv for interlock c

call Bob Smith x562 for PEO * NOTES: First thing in morning, contact Randy Wilson
arrange dedicated phone line for control Rm.
is for them to activate one of the spare ph
in the back room (there is a candidate rig
the corner from the computer room door, behi
cart holding a printer with a stack of label
run a line over the computer room door
blocked open door by the Shift Tech, in to
then down to the floor and across to the
is necessary to eliminate the distraction to
staff when we use the console phone n
the STA station

Also, per Mike Jensen, the FME workbooks fo
Jim Johnson (W) and Alan Joyce (W) have been
completed and NWTIMS will be updated 1st thi

Swingshift crew watched the top nozzle video
Seabrook. Recommend that all RE support
only a few minutes long)

2300 Randy Wilson / Jerry Mulligan in CR to set up phone
ops is being disrupted by using console phone f
communications. when they finish they will
problems of individual ericsson phones.

2230 (LE) Batteries in 2 CTMT phones were exchange
fresh batteries. Note that when on party line,
only last ~3 hours.

2305 Bob Smith called to tell us that SFP bridge cann
have NA'd steps involving real fuel, so those
will need to be completed (JAC will speak

2305 Midnight shift On - Jeff Camp Angel
Mike Baldwin John

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0000 Status:

Communications – we just switched to the new the RE desk in the CR and to connect everyone.

Cross checks – SIGMA checks are in prog grippers will not pick up

3303A-2 Are Relal check was done in April and

3303A-1 Weekly SFP Bridge day shift but stop visits not. We will do dummy gets back from contain

3303C-1 FIS Interlock we

0030 The delay on the SIGMA gripper appears will take several hours to resolve. So sending the dummy back to the SFP 3303C-1 and then 3303A-1.

0230 Performed 3303C-1 and it needs a procedure to delete section E. The change is we

Performed 3303A-1 and the SFP Bridge w Called Bob Austin who is out at SFP now.

One stop stop called to say that 3210 A pre see RCRA tool check which is 3303 D an in April. But they can't find it.

Word from SIGMA is that they are start it apart.

300 Another problem was noted with 3303C SIGMA interlock checks. In section 4 simulated on the operator and you the operator. It should not go but first we thought it may be the same with the gripper (ie SIGMA doesn't kn but then we thought it could be a problem. We will call Jim Sampse that all 3 problems at once.

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0400 Called Jim Flaspre
(1) He suggests we look for the 3303D Form
(2) we talked about the failed seal on SIGMA and he noted the same problem (possibly week then he welded the bypass key the operator when it should not have we This could indicate a problem with t (gripper up) or GUN (gripper up disengag switches
(3) we asked him about deleting section 8 we decided to only delete the part the bypass so that we can avoid c

0500 Word is that the SFP Bridge crane will TOC but they never actually did anything to it

Looked in Kathy's AWO but did not find 3

0530 Signed off ~~3303A-1~~ 3303C-1 (Rev4 Ch2) FT
Also Signed off 3303A.1 - jog problem fixed.

0630 Word is that SIGMA problem may be a connector a wheel

0730 Trying to get copy of 3303C-1 Rev4 Ch2 from CDL - they can't fi

0800 Copy of (W) Turnover documenting some of 3303D tool provided and added to AWO covering those tool

0830 Watched Seabrook tape of failed top nozzle Clamps - All of Day shift reviewed.

0915 CDL - gave us approved copy of 3303C1, Rev4, Ch2 - to (W) so they can finish section 4 and cop Section 8 (this section changed in Ch2)

~~1030 SIGMA appears to be~~

1100 SIGMA had been believed to be OK + when do checks, failed gripper checks. Further checks being

1115 Completed Cell Blocker verification in SFP surveill Per SP. 31022. Cell blockers verified in proper in Region I. Also completed FME log reco in SFP area with H.P. (Dennis Regan). t turned over the log to R.E. (it resides in MS-99-0001(c). The only item logged previously removed pages still under is the VAMP and tie-wraps that hold

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5/13 24 PROJECT NAME _____ NOTEBC

1300 [B] in SFP noted by Chemistry as 2787 as of 1025 on 5/12/99.
SFP level is SAT at 43% (in accepta
All Fuel Bldg. door checks required b
have been made and are SAT.

1300 SIGMA checks have been completed - (W) Mobi
on Station w/ copies of Refuel Work list
lights - to obtain better visibility in cavity.

All day shift personnel briefed for offload

1500 Swingshift on:

K Doroski, S Stack, K Connor, P Racicot, D A

Status - Were very close to starting core offload wh
on the Sipping console went out. (W) troub
found a dead circuit which appears to be
ground on the outlet (house power supply).
being sent in

Rx cavity water clarity is not too good.
is taking action to install a second Trip
Polar Crane (Oncoming W requested that water clar

Instructions for backup communications (Cas
been placed in the Fuel bldg, CONT and co
locations of pre-staged equipment

1630 SIGMA is downpowered due to electrical problem
is on his way into CONT to investigate. Replac

1843 Commenced Fuel Movement

1849 Engaged first fuel assembly.

1862 Override key for the crane is with t

1917 Overload on assembly G64 - Trip at weig

1919 Jim Gregan is on-shift SRO on bridge (SIGMA) (was b

1945 Sigma machine cannot release bundle. A S
will be checking overload situation

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5/13 25 PROJECT NAME _____ NOTEBOOK

* NOTE: Develop list of expected weights and place in Con-
: Address in EN31007 – provide instruction for moving
the core ~~out of the~~ out of sequence w/o sequence des

1946 Changed sequence on Refueling worklist to
pressure on baffle assembly G64.

2029 R-9 unlatched (FA G64) ~~unlatched~~ @ core

2058 Overload on H15 @ R-9 in core

* NOTE: Per Jim Beaupre, previous Rx Cavity water clarity c
due to poor lighting - additional lights were installed an
wattage bulb was replaced with a higher watt bulb. Cl
better.

* Late entry - Mike Jensen briefed the operating shift @ ~

2125 H15 unlatched in upender, core offload continuing.

* NOTE: SFP bypass key for upender in CR @ desk (signed o

2232 Lost communications in Control Rm.

2238 Communications restored

5/13 2300 Midnight Shift On – ~~Staff~~ Camp Ass
Mike Baldwin Std.

2320 Briefed oncoming STO.

2340 Per RCSs in SFP, definite gap observed in FA G37 (now
This is a discharge FA

2356 SIGMA LOCKED WITH H-29 FROM
CALLED ANNOUNCEMENTS, REQUESTED
WAIT REMUNERATION.

5/14 0001 SIGMA RESTORED.

0130 Returned bypass keys #50 and 59 to CRS lock
I don't want them.

0230 The offload AUO is with W out on SIGMA. I ask
move it to the FME table.

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5/14 26 PROJECT NAME _____ NOTEBOO

0516 G-35 FACE 4 VISIBLE GAP NOTICED DURING EXAMINATION

0615 went out to SFP to reconcile FME log. It w good but there is no operative block on the I seem to remember there used to be o then I went to containment to reconcile log. Found a couple minor problems that w but the AWO for offload door not hav made to reconcile FME after every shift, guess it was required in containment.

while in containment the guys showed m problem with the upender reservoir. It bill over the floor. It has a float v a toilet that sticks. We either move the float valve or get permission fro to operate the isolation valve

By the way - the offload AWO is at the FME desk now.

0630 Day shift in K. Emmons, J Chunis, J. Ely, M. Novera, K Claude Dean, J. Stoddard Turnover Brief complet.

0730 Notified by J. Stoddard that Craig Yoder (W) has reviewe of the f/a's w/ minor gaps (G-35 and G37) - his op that these are normal + no problem. His horsa

Confirmed that day shift - shiftly checks have bee on SF Bridge Crane.

1005 UPENDER IN SFP STUCK IN V POSITION, DOES NOT GO DO

1024 PERMISSION GRANTED FROM SM STEVE LAWHEAD USE BYPASS KEY FOR upender. KEY NOT IN CONTAI OBTAINING KEY FROM SM AND DELIVERING TO CONT upender. J. Beaupre says wait on key - looking at problem

1045 Assembly H28 on SIGMA lowered down in core location R08 BUT NOT UNLATCHED. for verdict on SFP upender.

1119 44 f/a's off loaded at time of upender in SFP

1155 LOSS OF COMMUNICATIONS BETWEEN CR 9 All stati

1206 COMMUNICATIONS RE-ESTABLISHED BETWEEN CR 9 All

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1210 SFP UPENDER HORIZONTAL. · continuing to run chec. and traverse cart.

1215 Communications LOST - All ericsons system went. N Wilson (Communications) notified - problem resolved + Communications reestablished within ~5 mins - No movement in progress at the time. (# to reestablish is 6

1230 Cycled upender after getting bypass key - appear a torque switch was tripped due to drive chain being jogged a small amount - when upender was put in horizontal position switch reset, Pizlury dummy out of U1 to put in upender + test ow

1247 Permission secured to resume offload - upender checked out O

1317 Refuel SRO used Bypass key to get full down indication in Cont. upender FA H09.

1330 SFP Upender will not lower. Signal to traverse to A-7 but will not latch until SFP resolved. I Stop/Bempe/W notified by K

1500 Swings on, no fuel movement in progress, 46 FAs out of core

K Doroski w/ S. Stalk, K Connor, P Raccoat, D Asay

Status: SFP Upender torque switch being examined - Jim Beaupre Ferns @ (12) (Spikey) in attendance

1540 Estimate ~2 hours to adjust torque switch. Communications will be relaxed as soon as SIGMA is moved away from the core.

1545 Communications relaxed, D. McCracken notified.

1752 Established communications with the Cont upender + SFP for some transfer system testing. Waiting for management to commence.

1830 Logboard verified.

1905 Mike Martell given pre-job brief as SIGMA SRO.

1950 Frame horizontal on upender - could not send to Cont side. Pushed in on hand wheel · cart tranferred successful

2009 Commenced fuel movement at core location A-7

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(NOT)

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5/14 29 | PROJECT NAME _____ NOTEBOOK NO

2022 Sigma machine having a problem unlatching in the upender. (W) going out to troubleshoot

2033 Sigma is going to bypass weight (take weight off) Unsuccessful. Troubleshooting other options.

2039 I+C contacted and going up to Cont to help with Sigma

2047 Sigma has indication problems both latched + unlatch lights on panel lit. They are going to hand crank to 800 lbs because they believe they maybe unlat I&C contacted to bring up tape or sleeving because may be a repeat of a circuit problem

2053 Fuel Assembly G74 is unlatched in the upender.

2205 I&C has control of Sigma - Handdown for 1 hour.

2300 SM concerned about rate of SFP heatup - a trend was generated (K:/i REGROOT/MISC/SEPTOMP.XLS) from IPCNT data on points SFC SFC-T18. He requested that it be updated periodically if tra permits.

5/14 2300 Mds crew on Jeff Camp Angelo Leone
Mike Baldwin John Magynik

5/15 0100 Just back from containment
SIGMA - I&C has pulled spare connector off spare mast. They are now soldering on. There are two connectors with wires each. They are still working the first connector when complet will check it out on the transfe canal fixture, then move fuel

these are the same connector that caused the problem 2 nights ago

Containment Upender - the problem noted la night with water flowing out the reservoir has been fixed. The valve 3CCP-V854 has been CAT tagged to say "float valve leaki open valve to fill tank" or some like that. The valve is now shi so the leak has stopped. I tol about this (they didn't know)

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asked them to check the level in the reservoir at every shift change. Cr
If the reservoir needs filling, we need tell the CR so we can get OAP to 3CCP-V854. If leakage out the reservoir had the thing would not keep filling at all.

SFP Upender - Once fuel movement starts a we will probably need to use the bypass key to lower the upender. plans to do some checks on it in operation. There are two possi problems (1) is the torque switch is so it doesn't stop the cart all the way in SFP. This theory has the switch work OK at first so the will lift, but while lifting it st and the torque switch kills con (2) the second theory is that the torque switch is OK and the really just all the way against stop. Some people think the holding the cart at full trave stick, or 1 is stuck.

Since the phase at the containment upender c causing problems, I took out a new ada and replaced it. That is the headset to pho plug in adapter at the bottom of the phone piece seems to me to be the weak link system.

0430 A meeting was held at the stop stop on SIGMA. had been tested after repair and would still no properly. I did not attend the meeting. It to get a straight story as to what the problem think the people are getting tired of being a one thing is for sure, they still think it a p with the connector.

0500 Checked the FME log on the SFP side OK. There is a large crowd of people heading into containment to work on SIGMA.

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0630 Day Shift on - Kathy, Kent, John Stoddard, Mark, Jay + C

0915 SF Bridge Crane Checks have been performed on day shift

0925 EN31013 has been completed SAT

0940 All communications verified established - Ctmt Upender, SFB Upender, SFB RE, SIGMA Operator, SRO in SIGMA

Late Entry 0930 SIGMA cable repairs complete per S. Brown - All necess retests complete per S. Koeser / S Brown

0945 Received permission from SM + Outage Director to resume core offload

0950 Latched assy H14 from core loc B7 to Decommin core offload.

1020 SFP Upender needed bypass, verified F/A out, received SM permiss

1115 Sigma will not reinitialize, notified 1 stop, W, advised SM that ref is stopped.

1125 Recommenced Retrieval, Sigma reinitialized

Late Entry 1200 SFP Upender needed bypass key on Step 114, verified empty, received sm Slight gap observed between face 3 and 4 of f/a H84 - Craig Yoder evaluating. Cart on SFP side is moving farther than should Jim Bergse observed + evaluating.

1402 STEP 118. Empty UPENDER WOULD NOT LOWER. SM GRANTED PERMIS TO USE BYPASS KEY to lower.

1402 STEP 119 SIGMA Full Down, RECEIVED FAULT ON PANEL, WOULD NO ENGAGE F/A. RAISED THE MAST AND CAME BACK DOWN ON UNSUCCESSFUL. TRIED AGAIN AND RECEIVED INTERMEDIATE GRAPPLE. WAS ABLE to CAME OFF F/A. INFORMED SM. SIGMA REPAIR PERSON NOTIF All Stop.

1420 SIGMA DID GRAPPLE ON TEST FIXTURE. FEELING IS IT WA NOT A SIGMA PROBLEM. SM GAVE PERMISSION to RETU to C5 AND try Again. - C5 grappled OK.

1425 Update on f/a H84 - separation has been measured at 31 mils - acceptable is 40 mils max

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1440 STEP 119, F/A HSS while going into upender Cl'lo
BOTTOM) lost Bottom, Open indication. RAISED P/A A
gripper indication. INPOWERED SM. Able to lower and
slack cable. Asked for & received permission to ge
Bypass to disengage. General Bypass did not work
F/A Finally unlatched in upender. F/A will be put A
SFP. All work on SIGMA is stopped concern of work
of procedure to unlatch. W also noticed thinnie plu
unlatch lit. Unlatch pushed and F/A disengaged.

1510 SM halted work due to questionable Contai
Isolation Valve (containment integrity).
Steve Lozier and Dennis Burton are stand
to troubleshoot SIGMA.

1530 SFP Crane Checks complete and SF

1625 Ctmt integrity restored

1636 Sigma Mast repaired and ready to commence fuel
offload

1640 Permission granted to commence fuel offload.

1645 Permission granted by SM to switch sipping rig to its own compres

1656 Latched in A-11

1703 Fuse blew on sipping machine compressor. Also lost SIGMA
1719 SM authorized sipping with N2 so that F/A can be
onto transfer machine. There is some concern that SI
pressure will bleed off before the blown fuse can be

1739 Unlatch F/A in upender. Moved SIGMA off upender to se

1746 Fuel movement suspended until SIGMA fuse is replaced.
down powered.

1750 SFP upender will not power. SM authorized use of byp
1755 Frame is lowered and key removed

1820 FUSE REPLACED ON SIGMA BRIDGE, COMPRESSOR OPERATING
TEMPORARY COMPRESSOR ON SIPPER POWERED DOWN. SIPPING
USING N2 BOTTLE. COMMENCING FUEL MOVEMENT AT THIS TI

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5/15/99
1856 Latched onto B-11

1900 "Hoist slippage" error on SIGMA. Proceeding with fuel hoist. SIGMA expert does not think the problem is significant.

1907 Unlatche F/A in upender

1917 SFP upender will not lower. SM granted permission to bypass lowered frame and returned to Reactor side.

2043 SFP upender will not lower. SM granted permission to bypass lowered frame & returned to cavity

2116 SFP upender will not lower. SM granted permission to bypass

2136 SFP upender will not lower. SM granted permission to bypass interlock.

2153 SFP upender will not lower SM granted permission to bypass interlock.

2209 SFP upender will not lower. SM gave permission to bypass interlock

2230 CONTAINMENT SIGMA CRANE COMPUTER SHOWING ILLOGICAL SETTING OF INFORMATION; COMPUTER REINITIALIZED AND FUNCTIONING NORMALLY

Date Entry Oncoming shift was briefed to maintain cognizance of requirements of fuel

2230 Plot of SFC-L26 & T18 was updated.

2240 There are 75 FAs out of the core

2245 SFP upender will not lower. SM granted permission to bypass interlock IAW OP3303C Precaution 3.2.2.

5-15-99 2300 Midnight shift on Jeff Camp Angelo
Mike Baldwin John

2302 SFP upender will not lower. SM gave permission

2334 SFP upender will not lower. SM gave permission to

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2350 Shiftly Pre-Op check of the Spent Fuel Bridge OP 3303A para 4.1 completed.

Checked with containment upender operator to ask how often they check the water reservoir level, they do every time a new operator goes on.

5/16 0005 SFP upender will not go down. SM gave permission to by

0055 SFP UPENDER WILL NOT GO DOWN SM GAVE PERMISSION TO BYP

0128 SFP UPENDER WILL NOT GO DOWN, SM GAVE PERMISSION TO BYPA

0130 Made a tour of SFP & Cont. In-mast sipping is o[illegible] off of N2 bottle. Total is ~ 1200 lbs which is good [illegible] ~ 12 FAT. There are two full bottles on there a[illegible] spares. FAT are moving well but the SFP is the weak link. The camera inspectors and the need [illegible] bypass on the upender about every other move is making SIGMA wait. Maybe the SFP is getting [illegible] with SIGMA for last night.

Reviewed FME logs in SFP and containment OK

0142 ASS. H-38 IS ~~[illegible]~~ BOWED AND SIGMA HAVING DIFFIC[illegible] PUTTING INTO UPENDER

0150 SM GAVE PERMISSION TO SIGMA TO USE BYPASS. WEIGHT + BYPASSED. ASS. H-38 DISENGAGED IN UPENDER.

0204 SFP Upender will not go down, SM Gave Permission

0205 Sigma ~~[illegible]~~ over core location J-9 + will not give cable indication. SM gave permission to Bypass Sigma's Height and Weight Interlocks to raise mast in an attempt to re-initialize memory. After raising and lowering mast, slack cable indication could not be established. Sigma was moved to Load Test Station awaiting assistance. Notified SM One Stop Shop and Refuel Team Lead. I+C and Wes[illegible] were contacted to investigate.

0245 Sigma repair team arrived

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0249 Re-initialized sigma - attempt to latch at core location 1-9

0255 Attempt to latch assembly successful.

0325 SIGMA had a problem latching the next FA also, but the experts got the thing working again.

Requested electrical maintenance to go to SFP to listen to the motor on the Bridge Crane. It's getting noisy.

0345 Electrical Maintenance says the noise from the Bridge Crane is not due to a motor problem but the cable on the drum. SFP Bridge Crane is being operated in slow speed only to alleviate this.

0350 SFP upender will not go down. SM gave permission to bypass

0405 SFP CRANE PICKING UP DUMMY to check the cable drum. Electrical maintenance noted a problem with the chain which drives the drum. This problem only occurs when operated in high speed. The dummy was never picked up the chain slips.

0415 SM gave permission to move the FA from the upender to SFP using slow speed. There is no FA latched in containment now.

0430 SFP UPENDER WILL NOT GO DOWN. SM GAVE PERMISSION TO BYPASS

0450 One Stop Shop had another meeting on fuel handling problems (getting to be a nightly affair). The chain causing the problem is to the hand crank which is what caused a problem last outage. They decided to first try to tighten it. Then it necessary do a temp mod to get rid of it.

0515 Fuel handling tool was placed in holder to support inspection of refueling crane

0536 Inspection of refueling crane is under way using basket to lift mechanics to hoist.

0613 SFP CRANE REPAIR COMPLETED. WILL HANG TO RUN IT AT SLOW SPEED FOR OVER TWO HOURS, WITH REPAIRMENT FOR THIS

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0630 maintenance has done a temporary fix to the d
wheel. They say we can continue moving fuel
slow speed until the Temp mod to remove it is
they poured some pine in the shaft and put l
on it. I have asked everyone to main sta
to start moving fuel again.

0700 DAY SHIFT ON- MARK, CLAUDIA, JOHN, KENT - LE
CATHY. BRIEF COMPLETED.

0705 FUEL MOVEMENT COMMENCED. PERMISSION RECEIV.
FROM SHIFT MANAGER.

0726 SPENT FUEL POOL CRANE SHIFTLY CHECKS COMP

0738 SFP UPENDER COULD NOT GO DOWN TO HORIZONT.
PERMISSION GRANTED FROM SHIFT MANAGER TO USE BY
KEY. UPENDER LOWERED AND TAKEN OUT OF B

0739 BLANKET PERMISSION GRANTED TO USE BYPASS KEY
SFP UPENDER TO LOWER IT FROM SHIFT MANAGER
UNDER THE CONDITION THAT WE CONFIRM THAT
IT CONTAINS NO F.A. AND THAT WE LOG USE O.

0800 SFP BRIDLE "SLIPPING" AFTER PICKING UP F.A. G-56
PERMISSION GRANTED FROM SHIFT MANAGER TO RETURN
G-56 TO UPENDER AND UNLATCH

0801 PERMISSION GRANTED FROM SHIFT MANAGER TO LOWER
G-56 IN SFP UPENDER IN FAST SPEED.

0801 PERMISSION GRANTED FROM SHIFT MANAGER FOR SIG.
TO TAKE F.A. H-77 TO CORE LOCATION R-8
DO NOT UNLATCH. F.A. JUST ABOVE CORE PLATE.

0802 SFP CRANE WILL NOT RAISE OFF FUEL ASSEMBLY.

0815 ON LOWERING F.A. H-77, BRAKE ON SIGMA WAS
NOT WORKING PROPERLY. SIGMA SRO (JIM KULL
WANTS TO WAIT AT ~ 1 FT OFF LOWER COR
plate and HAVE MAINT LOOK AT IT.
(DENNIS BARTON PAR REP)

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0826 SHIFT MANAGER DIRECTED THAT SIGMA BRIDGE IN CMT IS NOT TO LATCH ONTO 19 FUEL ASSEMBLY IN THE REACTOR VESSEL UNTIL THE FUEL ASSEMBLY IN THE SFP HAS BEEN REMOVED CLEAR FROM THE SFP UPENDER.

0833 ADJUSTMENT MADE TO SIGMA BRAKE, TESTING IN DOWN DIRECTION.

0835 BRAKE HAS BEEN REPAIRED, F.A. H-77 BEING LOWERED TO JUST ABOVE CORE PLATE. - CORE LOC R-8

0839 PERMISSION GRANTED TO LOWER SFB CRANE ONTO F/A BUT NOT TO GRAPPLE.

0846 HANGING SPANT FOR HANDLING TOOL IN STORAGE LOCATION.

0850 SFP UPENDER WILL NOT LOWER (CONTINUE F/A)

0914 SEQUENCE DEVIATION PERFORMED TO ALLOW PLACEMENT OF F/A H-77 TO CORE LOCATION A-8, SHAFFRONICS NOT UPDATED TO REFLECT DEVIATION

0941 F/A H77 IS DOWN AND UNLATCHED IN A-8 PER SEQUENCE DEVIATION. SIGMA WILL BE SECURED AND CORE ALTS HAVE STOPPED. ALL PERSONNEL CAN SECURE WORK LOCATIONS EXCEPT FOR SFP CRANE OPERATOR WHO NEEDS TO MAINTAIN COMMUNICATION WITH CR WHILE F/A IS VERTICAL IN UPENDER.

0950 SIGMA BRIDGE IS SECURE OVER UPENDER. PERSONNEL LEAVING STATION.

1225 ERICSON COMMUNICATION LOST ~ 1 MINUTE - SYSTEM PERTU-

1245 WORK ON SFP BRIDGE COMPLETE, TAGS BEING CLEARED. TEMP MOD TO REMOVE CHAIN FOR MANUAL CRANK REMOVED.

1258 PERFORMING WEEKLY CHECKS OF SF BRIDGE, PER AUTHOR OF SHIFT MANAGER.

1334 Weekly SF Bridge checks complete

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1337 VERIFIED WE HAVE COMMUNICATIONS WITH ALL WORK LOCATN. Shift Manager grants Authorization to commence core Alterations. After G56 has been removed from SFP upend.

1342 G56 latched in SFP. Fuel movement has recommenced

1344 By Pass Key used to lower SFP upender

1401 F/A H77 ungrappled in upender. This completes sequence deviation, Page 25 Step 181

LE 0730 F/A D51 Found to have estimated gap between top spring block & Top nozzle of 20-40 mils. This is a Discharge Bundle.

1410 Bypass Key used to lower SFP upender

1427 Bypass Key used to lower SFP upender

1429 SIGMA getting intermittent indications. SRO thinks pass water could be in air line. Dennis Barton called to D. air line. No impact to fuel movement

1446 Bypass Key used to lower SFP upender

1502 SIGMA crew draining mast

1505 Authorized PIT to put interlock control on upender into bypass to lower.

1515 Shifty pre-op checks completed on spent fuel bridge OP 3303A para. 4.1 compliance, notified control room SM

1530 G-10 unlatched in cement

1540 SFP upender would not raise with F/A H37. Sent ~~back towards cmt several feet~~. System Engineer is going out to look at it.

1558 SM gives SFP permission to move cart towards cmt several feet to troubleshoot upender. - On second attempt mech. switch made up and they raised the basket. The FA was and placed into the SFP

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1625 SUSPENDED FUEL MOVEMENT OPERATIONS AWAITING REPAIR OF SFP UPENDER TORQUE SWITCHES. ALL PERSONNEL TO BE AVAILABLE ON 15 MINUTE NOTIFICATION FOLLOWING REPAIRS.

1640 SFP Bridge crane tool off the hook and hung up for the duration. Preps being made to evaluate cause of upender problems. If no mechanical blockage found to explain fact that the cart is not making it to the stop, then they (Chris Ferris / Jim Beaupre) will attempt to adjust the torque switch.

1840 TESTING OF TRANSFER CART HAS PROCEEDED AND HAVE TRANSFERRED CARTS SEVERAL TIMES THROUGH TRANSFER CANAL TO RX BLDG AND SFP WITHOUT PROBLEMS. TESTING CONTINUES WITH DUMMY BUNDLE BEING SET TO PLACE IN UPENDER AND TEST CART WITH FULL WEIGHT LOAD.

1920 SFP CRANE LATCHING ONTO DUMMY FUEL ASSEMBLY IN "UNIFORM-1" TO TEST RUN UPENDER UNDER LOAD.

1924 SM RELEASING WORK PACKAGE TO TEST DETECTORS WEEKLY SURVEILLANCE; REFUELING WILL NOT BE ABLE TO RESUME UNTIL THIS IS COMPLETED BY I&C.

2010 SFP will be returning dummy in to SFP location "U"

2024 The "Dummy" FA is unlatched in SFP location "U"

2031 SFP upender is moving transfer cart back + forth to inspect.

2040 Aaron came to CR to brief REs & Shift on what the problem is with the transfer cart. Resolution is to use camera to monitor cart as it approaches stop @ SFP side. If "tab" does not deflect bar, then will back cart off a few inches and try again. SFP people have been trained what to look for using camera. Roger VanWey has been asked to initiate a change to OP 3302C to add clarification that if contact at hard stop is not obtained (as required by procedure right now), then back up and try again. Do not need the proc. change prior to moving fuel per J. Beaupre and SM (Ray Martin).

2101 Permission from SM to resume core offload

2117 Sigma put FA #G15 into upender but does not have indication that is down. Will be raising back up to inspect those indications.

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2127 FA G-15 Unlatched in CONT upender.

2245 99 FA's out of core

2250 There is a god... 100 FAs out of core

5/16 2300 Mids crew on. Jeff Camp, Kent Wiethorn, Angelo Leonel, Mike Baldwin

5/17 0000 Completed shiftly check of SFP bridge crane per OP 55 para 4.1. Asked for some status from [illegible] various things ① containment upender water reservoir still being checked with every new operator, level is st ② on both the rigging is 1100 psi ③ rigging graphs and going back to the trailer, no indications yet.

5/17 0020 F/A G69 (discharge) has loose holddown spring no while attempting to latch in SFP upender. S had no problems engaging this FA. SFP cr unable to latch at this time. Aaron Ellis his way to SFP. Camera in SFP Transfer Canal moved closer to get a better look at top no

0125 SM NOTIFIED F/A IS IN UPENDER IN ~~HORIZONTAL~~ vertical POSITION AND IS MANNED. We do not want to leave the upender in case the plate might fall out.

0145 We just had our nightly tool handling meeting at 0 and stop steps. We decided to modify the plant handling tool. I remembered we have a spare. Jim Beaupre was called, he says the spare 4 feet too short (from a plant with a different arrangement).

0330 Viewed the working end of the plant tool handling tool with the lightweight camera. It looks li the tool mod can be done without cutting or gri which makes things a lot easier.

0430 Finished our HP briefing on the tool repair. They going out now to start out a repair area to p it will be over the NEV. Thing RE needs to are ① with the new the camera equip ② clear Rick to move FA G69 to SFP NIP (versus N17)

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is low danger of dropping debris when moving the FA.
③ move camera support bar across transfer canal.
W Craig Yoder says there is little danger of any debris falling out of FA when it is lowered in upender, so that is what we will do.

0500 SM GAVE PERMISSION TO LOWER UPENDER ON SFP SIDE

0503 UPENDER IN HORIZONTAL POSITION ON SFP SIDE

0700 Turnover containment and SFP. Reviewed FME logs, had minor comment

0727 SM GAVE PERMISSION to BREAK COMMUNICATIONS BETWEEN CR & SFP RE AND UPENDER. RE WILL MAINTAIN COVERAGE AT SFP AND COMMUNICATE through NORMAL HOUSE PHONES.

1030 USING SFP CASK CRANE TO PICK SFP HANDLING TOOL. THIS REQUIRES ENTRY INTO T/S 3.9.12. NO LOADS OVERHEAD ON FUEL MOVEMENT WHILE CASK DOOR IS OPEN.

1045 HR BRIEFING TO PERFORM TOOL MOD - LOCATION HR CONTROL POINT.

1110 OPS CLEARING TAG ON SFP CASK DOOR
AWO'S ACCEPTED M3-96-5309
M3-99-5302

1230 CATCHER SFP HANDLING TOOL - BRIDGE CRANE

1245 OPENING ROLL UP DOOR - CMC ROOM NOTIFIED

1305 TOOL ON CASK CRANE HOOK.

1315 LARGE CASK CRANE HOOK WON'T GO HIGH ENOUGH

[illegible] INFORMED THAT ERICSON PHONE SYSTEM HAS BAD CPU. NEW ONE COMING w/ TECHNICIAN. MAY NEED TO USE ALTERNATE COMMUNICATION SYSTEM

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1410 (W) BRIEF - BOOST CONCIT IS TO GRAPPLE DUMMY "C" TIMES NEED SM PERMISSION TO GRAPPLE DUMMY AND 669

PERM TO PROCEED & MUST VERIFY PROC ORIENTATION.

1445 CAN'T GET TOOL OUT OF WATER IN VERTICAL, GOING TO USE BRIDGE CRA AND CASE MANIPULATOR PICK AND WAI ON WHILE SUSPENDED.

1450 TOOL HEAD OUT OF WATER

1500 BOTH P.N & STOP BOARDS - PUTTING TO IN WATER

1520 SAFETY CRANE CHECKS PERFORMED IN SFP.

1530 BOOST ORIENTATION BEGUN SFP COOLING T AREA 1800 gpm

ROLLUP DOOR CLOSED, TAGS HUNG NEW A BACK TO START WORK.

1545 EVERYONE ON STATION (2nd SHIFT) COMMUNICATI ESTABLISHED AUD'S IN CTMT @ FUEL DISC AND SFP BY VIDEO MONITOR.

WATER TEMP MOD PAGER TO FINISH SFP BOOST TESTING OF TOOL & FUEL MOTION

1615 SFP TOOL BOOST COMPLETE And Temp Mod signed

1630 UNCAPPED P/A 669

1641 P/A 669 IN SFP LOCATION N-17

1657 BYPASS KEY UTILIZED TO LOWER UPENDER AT SFP TRANS

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* NOTE - change in "process" - SM has given permission to latch in Rx prior to clearing basket in SFP BUT can't raise FA in Rx until SFP side basket is clear.

* NOTE - SM (steve lawhead) has given permission to the Lead RE to allow bypass of SFP upender (IAW OP3303C Step 3.2.2) without checking in with him each time. This may change when the next SM comes on.

* NOTE - Jay Ely performed a review of all of our procedures as well as the SAR and verified that the alignment pin which was removed from the spent fuel handling tool is not credited anywhere.

1753 SIGMA BRIDGE UNABLE TO GET ENGAGEMENT LIGHT AFTER (4) FOUR ATTEMPTS TO LATCH ONTO FA "H-24" AT CORE LOCATION "C-12".

1905 SUSPENDING REFUELING OPERATIONS TO ALLOW FOR REPAIR OF SIGMA BRIDGE BY I&C. UPENDER TO SPENT FUEL POOL SIDE. WESTINGHOUSE SIGMA OPERATOR REMAINING IN CTMNT ON HEADSET TO MONITOR PROGRESS OF REPAIRS.

2300 Midnight shift On Jeff Camp Angela Leone
Mike Baldwin Kent Christiansen

5/18/99 Completed shiftly check of SFP Bridge crane per OP3303A page 4.1.

0115 SIGMA Rx did not work. All personnel are relieved from their station. I&C went and did another check of the solder joints they did OK so it must be the connector itself.

Called our nightly refueling meeting in the air stop shop. We decided to try to get rid of the connector by using a butt splice. If that doesn't work then the entire cable will be replaced. Estimated time to get the butt splice in is 4 hrs.

0700 Repair of SIGMA cable is in progress.

0700 DAY SHIFT ON: KATHY (LEAD), MARK, JAY, CLAUDIA, JOH

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0745 SFP BRIDGE SHIFTLY CHECKS COMPLETE.

0747 COMMUNICATIONS ESTABLISHED WITH ALL STATIONS.

0747 BLANKET AUTHORIZATION RECEIVED FROM SHIFT MANAGER TO USE SFP UPENDER BYPASS KEY TO LOWER UPENDER AS LONG AS IT DOES NOT CONTAIN A FUEL ASSEMBLY

0754 SIGMA REPAIRED- AUTHORIZED TO PROCEED WITH RETEST OF SIGMA.

0757 REFUEL SRO CONFIRMED THEY HAVE AUDIBLE COUNTS IN CONTAINMENT.

0759 SIGMA RETEST COMPLETE SATISFACTORILY. AWAITING AWO RETURN TO OSS.

0800 CONFIRMED SHIFT MANAGER AUTHORIZATION TO LATCH, BUT NOT RAISE, CORE FUEL ASSEMBLY, WHEN A[illegible] FUEL ASSEMBLY IS IN THE UPENDER IN SFP. RAIS[illegible] OF THE CORE FUEL ASSEMBLY CAN NOT PROCEED UNTIL FUEL ASSEMBLY IN SFP UPENDER IS CLEAR OF THE UPENDER.

0810 AUTHORIZATION FROM SHIFT MANAGER RECEIVED TO PROCEED WITH CORE OFFLOAD.

0810 CONFIRMED WITH JIM BEAUPRE THAT IT IS PREFERRED TO USE DOUBLE BUMP OF SFP UPENDER EVERY TIME.

0813 RICH MADEROS NOW REFUEL SRO UNDER INSTRUCTION

0921 SIGMA COMPRESSOR DRAINED

1000 CONT INTEGRITY LOST- LOTE alt momentarily suspended

1040 Containment integrity restored- permission to resume co[illegible] offload given by SM. Communications check perfor[illegible] SAT.

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1145 SIGMA INDEXTION PROBLEMS - CAN'T ENGAGE. STEVE LOWER IN CONTACT WITH SIGMA FOR RESOLUTION. ALSO LOCATION INDICATION PROBLEMS SIGMA RE-INITIALIZED.

1205 SIGMA PROBLEMS CONTINUE. STEVE & DENNIS ARE GOING OUT TO SIGMA.

1248 SIGMA ENGAGED AT M12 AND MOVING. PROBLEMS APPEAR RESOLVED EQUIP REP FEELS THIS WAS A PROGRAMMING PROBLEM AND NOT A HARDWARE PROBLEM

1342 SFP UPENDER WILL NOT RAISE. DOUBLE BUMP JOG WAS PERFORMED. F/A H43 IS IN UPENDER. JIM BEAUPRE IS GOING OUT TO SFP. SM INFORMED.

1358 SFP UPENDER HAS FRAMED UP. FUEL MOVE CONTINUES.

1438 SIGMA CHANGING NO_2 BOTTLE

1455 SM (S. LAWHEAD) HAS GIVEN PERMISSION TO START RAISING FUEL FROM CORE WHEN UPWARD MOTION OF FUEL IN UPENDER @ SFP HAS BEGUN.

1500 SFP SHIFTY CRANE CHECKS COMPLETE & SATISFACTORY
~~Roll up door tag verification complete & satisfactory~~

1518 SFP upender will not raise F/A G-28 is in the upender Shift Manager gave permission to bleed the system

1621 Success - The frame is up in the pit. -

1635 Having difficulty placing F/A G28 in SFP Location B1 SUCCESS!! ☺

1649 Upender will not raise contains F/A H44 - SM gave permission to bleed the system.

1720 Upender will not raise - contains F/A G12 - SM gave permission to bleed system. Having problems making up the mechanical switch. Jogging back & forth continuing.

1757 Jim Beaupre notified & some options are being discussed.

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1830 FRAME IS UP @ SFP, FA G-12 WILL BE TRANSFERRED TO SFP AR

1835 RECOMMENDED STAND-DOWN UNTIL TROUBLESHOOTING OF TRANSFER SYSTEM IS COMPLETE AND CAUSE OF UPENDER PROBLEM IS UNDER FA G-12 IS IN SFP AR34. AWO REQUESTED FOR USING "LONG POLE" NECESSARY TO MANUALLY ACTIVATE THE MECHANICAL INTERLOCK

1940 TRANSFER SYSTEM HAS BEEN TESTED USING DUMMY FA AND IS OPER FLAWLESSLY. DISCUSSED WITH SM, GARY SWIDER, AND BARRY F THEY ALL CONCUR - WANT TO RESUME OFFLOAD. AWO STILL PROCESSED FOR "LONG POLE", AND POTENTIAL USE OF A DIVER BEING CONSIDERED.

2010 RESUMED OFFLOAD - RX LOCATION C-14

2011 Sigma needs reboot. ~~SFP~~ SRO reports that they are having problems with SIGMA not lining up with core location C-14. They are going to jog bridge & trolley then try again. Seems to be a core boundary problem according to Refueling SRO.

2016 Sigma is going down to core position C-14.

2045 ~~2017~~ ~~MAYBE NOT...~~ GRIPPER TROUBLE RECEIVED, APPARENTLY DUE TO A UNDERLOAD CONDITION. (W) TECHIE IS HEADING INTO COMT. TRANS CART SENT OVER TO SFP FOR TROUBLESHOOTING. MANAGEMENT NOTI D.

2143 STANDDOWN RECOMMENDED TO ALLOW I&C AND WESTINGHOUSE TO COMPLETE TESTING AND TROUBLESHOOTING OF SIGMA BRIDGE. A REFUELING CREWS STANDING DOWN. CONTAINMENT SRO OFF SIGMA BRIDGE

2300 Midnight Shift On — Jeff Camp, Mike Bolde; Kurt Wiatrom, Angelo Ler; Don Asay

5/19 0100 Jogging transfer cart 6-8" inches from full tr limit in SFP to check mechanical latch th signals cart fully in place. (torque switch che Next sent cart all the way to reactor side then back again. Aaron Ellis in SFP obse

0110 Received permission from Ray Martin to raise upender in SFP using by pass, since it wo not raise normally. Had run cart to full limit, but would not raise. Bypassed i but frame still would not raise.

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5/9	0115	Checked reservoir level (good) for upender and verified bleed valves were closed also. Sending transfer cart 8"-12" back toward Rx. and back again to make up latches/interlocks.
	0124	Sent to Containment but Upender would not raise to vertical there either. Sent back to SFP.
	0130	UPENDER IN SFP, STILL UNABLE TO RAISE
	0135	UPENDER SECURED IN SFP + OPERATORS SENT OFF STATION
	0230	SUCCESSFULLY RAISED UPENDER ON SFP SIDE. SIGMA UNDERGOING CABLE REPLACEMENT
	0430	SIGMA REPAIRS COMPLETE. SRO, (W), AND RE'S RETURNING TO STATIONS. SIGMA CHECKS ARE UNDERWAY.
	0456	SFP PLATFORM CRANE CHECKS COMPLETE. 3303 A 4.1
	0500	VERIFIED OFFLOAD/RELOAD AWO IS AT CONTAINMENT FME DESK. SFP COOLING FLOW IS 1800 GPM
	0510	I+C SAYS SIGMA IS READY TO GO
	0511	SM GIVES PERMISSION TO BEGIN MOVING FUEL CART BEING SENT TO CONTAINMENT
	0516	SIGMA OVER CORE LOCATION D-14 AND GOING DOWN
	0518	SIGMA LATCHED ON IN CORE LOCATION D-14 AND COMING UP
	0527	FA G-03 UNLATCHED IN CONTAINMENT UPENDER
	0538	FA G-03 CLEAR OF UPENDER ON SFP SIDE
	0544	FA G-03 IN SFP LOCATION P-5
	0615	SIGMA OVER CORE LOCATION N-14 BUT UNABLE TO GO DOWN. WILL TRY MOVING OFF N-14 ~~AND~~ TO H-8 THEN BACK TO N-14 TO ~~RETRY~~ RETRY GOING DOWN
	0617	SIGMA UNABLE TO GO DOWN ON CORE LOCATION N-14
	0620	SIGMA GOING DOWN ON CORE LOCATION N-14. HAD TO MOVE

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SIGMA SOUTH AND WEST, THEN ABLE TO GO DOWN. SUCCESSFULLY LATCHED ONTO ASSEMBLY

0655 AUTHORIZATION RECEIVED FROM SHIFT MANAGER TO LATCH & RAISE F.A. IN CORE AS SOON AS F.A. IN SFP UPENDER HAS BEEN LATCHED AND IS BEING RAISED.

0700 DAY SHIFT ON - KATHEY, JOHN, CLAUDIA, John Magyarik

LE 0690 Reviewed FME log in SFP and bypass key log in. Jon office FME said we have 3 keys, Jon log sheet 4 keys. Forced the upender operator who says he had 4 keys. Corrected FME log.

0732 SFP shifty checks on Bridge complete

0754 SFP upender lowered a light in transfer canal to see switches better.

0757 SFP upender bleeding value.

0809 Rich Maderios on SIGMA UNDER INSTRUCTION.

0839 SFP UPENDER VENTING SYSTEM FOR F.A. D76. SIGNIFICANT PROBLEM THIS TIME WITH UPENDER, SEVERAL ATTEMPTS WERE NECESSARY TO RAISE IT.

0857 SIGMA DRAINING COMPRESSOR AND AIR HOSES.

0900 SFP UPENDER BLEEDING SYSTEM.

0900 LATE ENTRY - OPS INITIATED MAKEUP TO SFP ~0800.

0911 MAST HUNG UP ON ROLLER BEARING - NOW FIXED AND WORKING OK.

0900 Temp tag hung on SFP upender ~~vent~~ valve for bleeding off sys + independently verified by J. Beaupre

0950 SFP UPENDER BLEEDING SYSTEM.

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1000 Verified SFP Bridge Weekly Crane Checks performed - OP 3303 A-1 and SP3603A1.1 5/18/99

1000 Jack Johannsen has another phone programmed into party line - next time we stop, he will tie into the system. Let him know when we stop again. He is at X 6543 when we are ready to have him tie phone in.

1015 AUTHORIZATION FROM SHIFT MANAGER TO USE SFP BRIDGE BYPASS KEY TO RAISE F.A. FROM N-3 AND RE-LOWER.

1020 SFP BRIDGE BYPASS RETURNED TO NORMAL.

1024 SFP UPENDER BLEEDING SYSTEM

1039 CTMT UPENDER REPORTED THAT H63 BOWED PRETTY BAD.

1045 SFP UPENDER AUTHORIZED TO BLEED SYSTEM.

1047 SFP UPENDER AUTHORIZED TO BLEED SYSTEM A SECOND TIME.

1113 SFP UPENDER Bleeding system. HAD TO BLEED SYSTEM 3 TIMES.

1130 RECEIVED DATA FROM OPS: BORON CONCENTRATION.

RCS [B] 2776 PPM @ 0840

SFP [B] 2770 PPM @ 1100

1132 SFP UPENDER Bleeding system. HAD TO BLEED 4 TIMES.

1158 SFP UPENDER Bleeding system. (one Bleed)

1215 SFP UPENDER Bleeding system

1216 RICH MEDIROS NO LONGER ON SIGMA under INSTRUCTION.

1240 SFP Upender Bleeding System

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1256 SFP Upender Bleeding system (Three attempts before able to raise frame)

1318 SFP UPENDER BLEEDING SYSTEM (ONCE)

1403 SFP UPENDER BLEEDING SYSTEM F.A. G24 (2 TIMES)

1411 SFP RE REPORTED THAT FUEL ASSEMBLY G24 HAS A SLIGHT CRACK ON SPRING BLOCK MATING FACE, DEFINATELY HIGHER ON ONE SIDE. NEEDS FURTHER W/ EVALUATION. (E) Evaluation determined No observable damage.

1422 SFP UPENDER AUTHORIZED TO BLEED SYSTEM F.A. H21

1426 JIM BEAUPRE WORKING ON SFP UPENDER LATCH.

1433 UPENDER TROUBLE shooting complete.

1450 SFP UPENDER AUTHORIZED TO Bleed SYSTEM F.A. H71 (4 TIMES)

1500 SWING SHIFT ON STATION: K DOROSKI, P RACICOT, S. STARK, K CONNOR

1510 SHIFTY CRANE CHECKS COMPLETE

1520 SFP UPENDER AUTHORIZED TO BLEED SYSTEM F.A. G-13 (4 TIMES) PRIOR TO UPENDER RISING

1523 BYPASS KEY REQUIRED TO LOWER UPENDER FRAME W STORE FUEL POOL PIT.

1550 SFP UPENDER AUTHORIZED TO BLEED SYSTEM TO ALLOW RAISING OF UPENDER FRAME WITH F.A. D-75. (4 TIMES)

1608 SFP UPENDER AUTHORIZED TO BLEED SYSTEM TO ALLOW RAISING OF UPENDER FRAME WITH FA G-25 (2 TIMES)

1625 REINITIALIZED SIGMA BRIDGE AFTER UNLATCHING F.A. H-20 AND PROCEEDING TO L-3 LOCATION.

1635 SFP UPENDER AUTHORIZED TO BLEED SYSTEM WITH VENT VALVE TO ALLOW RAISING OF UPENDER FRAME WITH F.A H-20. SFP UPENDER OPERATOR VENTING VALVE TO 5 GALLON POLY BOTTLE AT THIS TIME. (5 TIMES)

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1655 REINITIALIZED SIGMA BRIDGE CONTROLS AFTER UNLATCHING F.A. H-54 AT CONTAINMENT UPENDER.

1715 SFP UPENDER AUTHORIZED TO BLEED VENT VALVE FA H-54 (3 TIMES)

1740 SFP UPENDER AUTHORIZED TO BLEED VENT VALVE FA G-14 (5 TIMES)

1810 SFP UPENDER AUTHORIZED TO BLEED VENT VALVE FA D-84 (12 TIMES)

1850 SFP UPENDER AUTHORIZED TO BLEED VENT VALVE F.A G-06 (12 Times)

~~1710 Cart is going to home switch the switch spring loaded is not clearing the hardstop on the upender for H-04~~

1720 Will not be picking up Fuel Assembly ~~E2~~ in the core until we get someone to access the upender problems. Getting progressively worse.

Note Entry 1710 SFP RE reports that the cart is going to the home switch but the spring loaded ear is not clearing the hard stop

2003 SFP RE reported a black tie wrap was found on the track in the SFP transfer canal.

(W) MOBILIZING "LONG POLE" AND WILL PROCEED AS FOLLOWS:
- RETRIEVE TIE WRAP
- ACTUATE MECHANICAL INTERLOCK WITH POLE AND TEST FUNCTION
- CYCLE TRANSFER SYSTEM W/ DUMMY FA TO VERIFY PROPER FUNCTION
- WITH APPROVAL OF SM, RESUME CORE OFFLOAD

2105 TIE WRAP RETRIEVED AND PLACED IN BASKET

*NOTE - IF TRANSFER CANAL IS DRAINED DURING MODE 6, SHOULD INSPECT REMAINING TIE-WRAPS ON "ARMORED CABLE" FOR TRANSFER DRIVE.

2126 DUMMY FA ~~IN UPENDER~~ HEADING FOR UPENDER - DID NOT RAISE ON FIRST ATTEMPT - NEED TO MOVE TO HARD STOP, MANIPULATE INTERLOCK, BACK OFF, BACK IN, THEN UPEND.

2245 PLACED DFA IN CART, SENT TO CTMT, RETURNED TO SFP, JOGGED CART IN UNTIL "CART AT PIT" LIGHT LIT, THEN MANUALLY ACTUATED SWITCH W/ POLE. WHEN LEVER WAS PUSHED BACK, CART MOVED FORWARD A LITTLE DUE TO RESIDUAL TENSION ON CHAIN. CART UPENDED SUCCESSFULLY.

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2150 RECEIVED SM PERMISSION TO RESUME CORE OFFLOAD - PERFORMING PHONE CHECK.

2260 FA D79 INDICATED AS A LEAKER (DISCHARGE ASSEMBLY!)

2320 Shifty SFP Crane checks complete and SAT as required by 3303A Sec 4.1.

2340 In Mast Sipper Operator reports that signal from D79 indicated a small leak (500 counts). After sipping he did a purge for a couple minutes and then ? blank test for a total delay of about 15 min.

In my turnover from Swingshift I am told that the log entry from 1411 saying that FA G24 has a crack is incorrect.

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0120 FA D69 appears to have a damaged top grid strap on face 4. The entire grid strap appears tiny so we didn't tell if there new damage or not. Also can not tell if any part of the grid is missing. Face 4 was against barrel baffle. Aaron Ellis reviewed the tape a D69 and agreed with the above. Aaron recommend a close look at G57 which is the only face adjacent FA which has not been removed yet.

0230 FA G58 with the source dell not want to get into the core at location H15. Brought in additional lighting.

0303 FA G-58 DOWN IN CORE LOCATION H-15
Count on SR92 went from ~ 2 cpr to ~ 50 cpr.

0330 Requested Electrical Maintenance to bring additional lighting to the core. RPO says the reason for the delay in the G58 move was poor lighting.

LE 0100 Reviewed FME log in SFP. Found one minor discrepancy.

0500 SFP RSO reports SFP hoist "getting tough"

Elect Maint working with W to install light in Rx.

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0510 Switch audible count to channel 31 in prep for next source movement.

0530 The upender in SFP has behaved very well tonight. It occasionally needs to be "double bumped" to get full travel, but usually not. I think the "long pole" treatment on swing shift taught it a lesson in proper manners.

0615 Moving the second source went without any trouble. SR31 behaved just like SR32. Count rate went from 2 cpr to 50cpr.

0700 DAY SHIFT ON: JEFF, MARK, JOHN, CLAUDIA; Kathy

0725 Shiftly SFP crane checks complete per of 3103A para 4.1.

0800 OPINION OF SFP RE: UPENDER operator is that SFP CRANE HOIST SOUNDS the same as it always has, no louder than usual.

0902 F.A. 6-51 TRANSFER TO SFP complete, PAST TRANSFER CANAL GATE - SHIFT MANAGER DECLARED MODE Φ.

0945 SFP LATCHED ONTO TRANSFER BASKET AND LIFTING AT 267 LBS.

0955 TB-1 UNLATCHED IN SFP UPENDER.

0958 TB-1 GOING TO REACTOR.

1001 TB-1 IN CONTAINMENT, FRAME UP. AWAITING SIGMA PERSONNEL TURNOVER.

1152 DIRECTION GIVEN TO SHIFT MANAGER THAT IT IS ACCEPTABLE TO RETURN SFP COOLING TO NORMAL WHILE WE WAIT FOR TRANSFER BASKET TO BE LOADED IN CONTAINMENT. SFP COOLING WILL BE REDUCED WHEN BASKET IS BEING RETURNED TO THE SFP.

1221 SPECIMEN FREE - MOVING TOWARD CTMT UPENDER.

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12:29 SPECIMEN IN UPENDER IN CTMT.

12:34 SPECIMEN IN SFP.

12:40 BOTH PLUGS REPLACED IN VESSEL.

12:45 ATTEMPTING TO GRAPPLE TRANSFER BASKET IN SFP.

13:20 COMMUNICATION STATION IN CONTROL ROOM DE-STAFFED. SFP UPENDER & CTMT UPENDER WILL MAINTAIN THEIR OWN COMMUNICATIONS.

1455 Trash basket bottom is tweaked - finally came out - put in A1

1500 SHIELDED CRANE CHECKS DONE; SFP CB = 5/19 2770 PPM

1505 SWING SHIFT ON: K DROSKI, P RACICOT, K CONNOR

1545 WEIR GATE BEING RIGGED FOR INSTALLATION; (W) CAMERA OUT OF CANAL

1630 WEIR GATE IN; CAMERA BEING PLACED IN POOL, BOOT NOT PRESSURIZED

LATE ENTRY 1600 (W) CREW/RE CREW BRIEFED ON INSERT SHUFFLE EVOLUTION

1705 READY TO BEGIN INSERT SHUFFLE - SHIFT NOTIFIED
FUEL POOL COOLING INCREASED TO ~3400 GPM WITH THE UNDERSTANDING THAT IF CLARITY IS ADVERSELY AFFECTED, THEN THEY WILL THROTTLE BACK.

1740 REQUESTED OPS REDUCE SFP COOLING FLOW - THEY WILL DECREASE FLOW 3500 TO 2500 GPM. (TOO MUCH TURBULENCE)

* NOTE - IF ANY BREAK IN SHUFFLE MOVEMENT, NOTIFY OPS TO INCREASE COOLING FLOW IN SFP (PER THEIR REQUEST)

2200 HEALTH PHYSICS, IN CONTACT WITH CHEMISTRY, STATES VENT BOTTLE OF H_2O CLEAN FOR DISPOSAL DRUM DRAIN. HP WILL DISPOSE OF H_2O.

LATE ENTRY 1900 (W) CREW BRIEFED ON INSERT SHUFFLE

2200 (W) CREW BRIEFED ON INSERT SHUFFLE

2300 Mid-shift: K. Wiethorn, T. Camp., D. Asay

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*NOTE - OPs STARTED GOAT PURGE AND NOTICED LEVEL CHANGE IN RX CAVITY AND IN SFP. THEY NOTED THAT THEY HAD ~¼ TURN ON THE GATE IN BUT IF LEAKAGE IS NOTED AFTER DRAINDOWN, MAY WANT TO HAVE ENG EVALUATE FOR ADDITIONAL TORQUE ON VALVE.

2350 Shiftly SFP Crane checks completed and SAT per OP3303A para 4.1.

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

0515 While using the tie off cable which secures the RCCA tool to its bracket on the side of the SFP, one of the pipe clamps was observed to be loose. It was removed by hand and taken out of the FME area to see any parts were left. This cable needs to be fixed by maintenance.

0700 Reviewed FME Log in SFP, OK.

Night to Day

0630 Day shift here - Kathy, Claudia

0700 Briefed (W) Crew on insert shuffle. CBD briefed also.

0735 Shiftly SFP Crane checks complete and SAT. per OP3303A para. 4.1

1500 Completed Thimble moves and changing out tool to perform rod moves.

1515 SHIFTLY ~~BRIDGE~~ SFP CRANE CHECKS COMPLETE AND SAT PER OP 3303 A PARA. 4.1

1605 PROBLEM WITH RCCA TOOL - WERE NOT LATCHED AT U-12 (STEP R2) AND RAISED TOOL, WHICH MESSED UP THE TOOL "SEQUENCING". HAD TO HANG TOOL AND MANUALLY "RESET". R5 & 6 SUCCESSFULLY LATCHED AT 1700.

1850 HP wants to take camera and long pole out. (W) says leave camera, but pole can come out. Estimate ~10 gal clean water. CR notified - they said go ahead. FLOWMETER READS 3296.

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1835 (W) SWITCHING TO THIMBLE PLUG TOOL FOR LAST 2 MOVES.

1910 "LONG POLE" OUT OF TRANSFER CANAL

1910 LAST INSERT SHUFFLED, HANGING UP THIMBLE PLUG TOOL.

1915 CR NOTIFIED INSERT SHUFFLE COMPLETE, AND THEY CAN INCREASE SFP COOLING AS DESIRED.

AFTER RINSING POLE, WATER METER READS 3306.

5/26/99 1100 While working on communications gaitronics sys on SF Bridge, dropped wire nut into SFP

Retrieved wire nut from top of cell blocker in SFP at Location F2.

Drained xfer canal to work on transfer cart/gender problems. Replaced springs in cart. Worked on latch trip mechanism.

5/30/99 1100 Retrieved spring clamp and 3 screws from Assy 669 in SFP loc N17. Placed in wire bucket and transferred to shipping cask up in cask pit area.
Both AWO's are in 1 stop shop so tags can be lifted on SF cask area roll up door to get cask out + down to canopy area.

5/31/99 Shuffleworks is ready to go. Head cask w/ spring + clamp is in canopy area. One stop shop has our work orders for SFP + Offload/Reload. Need to get them re-released for us to work FOSAR of core plate. Call Rich Medeiros if fuel movement will NOT happen on midnight shift.

5/31/99 1530 Started inspection of upper internals pins. Kent is out there

1700 Upper internals pin inspection complete, no problems or damage noted

*Note - All video tapes have been placed in the horizontal file (top section) 2 AWOs re-released to MD - m3 96 05319 is in CTMT near HP @ 51' for FOSAR

1800 Went into CTMT to see if (W) ready for FOSAR - ladder in the way. Mate was called, they will remove shortly.

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1830 (W) requested additional filtration in Rx vessel; HP will try to set up another Tri-nuc. Also (W) requested (again) a dedicated power supply. OSS said they will try.

1900 KMD & KJW went into CTMT for FOSAR - (W) had just discovered a crack in the camera lens cover. They rigged it anyway and it was working fine until they set it up on SIGMA. It failed @ ~ 2115. We exited CTMT to let Aaron know what was going on. They will get camera from SFP to do FOSAR - expect to be ready around 2300.

*NOTE - make sure ALL required crane checks get (got) done.

5/31 2300 Midnight shift On - Jeff Cayer, Mike Baldwin, Angelo Leone, Don Hray

Things to do in no special order -
camera up both core plate
deck communications, locate panel, headsets
crane checks
set up work station with RWL's, AWOL, battery charger etc
cleanup Rx cavity water
other misc tasks

6/1 0500 This is the status:
I sent the crew off to work stuff on the list above while I went into containment. We were then checking out cameras & lights. HP and elect paint was installing the guide pin lights. We started the FOSAR at the lower core plate at about 0200. Found a wire or nail at location D5. The wire back afterwards with an Elmo camera to get a better look. Notified HP, DPI, FMGT deck, and Robert Tesu Coord (Roger Van Wey). The plan is to pick it up with a TRIDEC. While we were doing this DPI was restoring loops. A burst of air bubbles from restoring loop tripped a light up into our eyes. So after moving the lift out attaching a second rope to the TRIDEC hook we will pick it out.

DPI is just now opening the transfer tube gate valve.

The crew checked communications, located panel

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got to crane dock paper, looked out shuttle work. Other mite stuff.

Next is to rock out wall, then we will take some of their general back to the SFP, do SFP crane check with the dummy, and dummy to contain, then do SGMA check with dummy.

0530 Roger Van Wey volunteered to write the CR for the debris in the Rx. Roger also called Vince Spivan (Law ext 18) for an OC of the recovery plan.

The wire has been recovered. (wire or nail, whatever)

The lower core plate inspection and wire/nail recovery were taped. Sent Angelo out to get the tape.

0550 Angelo returned with the FOSAR tape.

Gary Swider says the gate valve is open and we are free to move the dummy.

0630 CURBO will not update since P3-CHRONOMETRIC is not running thanks to Bill Lanan, he is investigating.

Also - while looking at the SFPC,

CURBO is calculated using the following

note: for future the formula should be referenced, checking the constant must be done via AWO through computer access/need to check to ensure any changes are valid.

$$CURBO = \frac{CUINPO}{SCMTU}$$

where CUINPO is integrate power and SCMTU is the core loading in MTU. The constant in the system was 89.294 (A cycle) It has been changed to 87.774 (cycle 7 value from NDR pg 2-6)

0715 SFPA turnover C. Dorn, J. Cry, J. Chinn, M. Nacona

0815 EN31013 for moving Dummy performed

0815 Performed weekly SF Bridge checks + Shiftly checks

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0825 COMMUNICATIONS ESTABLISHED WITH SIGMA, PIT UPENDER AND CTMT UPENDER.

0827 LATCHED ON DUMMY DOING SFP CHECKS IN PIT

0829 UNLATCHED FROM DUMMY IN UPENDER IN PIT.

0835 AUTHORIZATION FROM SHIFT MANAGER TO SEND DUMMY TO CTMT.

0837 DUMMY IN CTMT.

0840 FRAME IN CTMT CHECKS OUT OK.

0859 SIGMA CHECKS complete. DUMMY MOVED TO PIT.

0915 DUMMY UNLATCHED in SFP LOCATION UNIFORM 1 (U-1)

0930 RELOAD PRE-JOB BRIEF HELD IN (W) trailer.

0930 CVRXBU is re-zeroed in PPC

~~0~~ 1030 Starting Background count rates for 1/m for reload sight's

1105 AUTHORIZATION RECEIVED FROM SM TO PLACE SHOE HORN ON H-1.

1108 GAMMAMETERS PLACED IN 1/m MODE.

1119 Communications ESTABLISHED BETWEEN All REFUELING stations

1122 SM AUTHORIZES PERMISSION TO COMMENCE CORE RELOAD

1.30 SFBridge operator proceeding to first SFP Location

1145 RE-INITIALIZING SIGMA prior to LATCHING 1ST F/A
SIGMA REQUEST DENNIS BARTON TO come out to SIGMA.

1215 FIRST F/A BROKE VOSSOR FUNCTI CORE RELOAD

1230 BLANKET PERMISSION FOR SIGMA BYPASS for PURPOSE of MOVING SHOE HORN.

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LATE 1130 ENTRY — SPENT FUEL BRIDGE BYPASS KEY #59 IS SIGNED OUT TO JOHN S. THIS KEY IS TO BE IN SPENT FUEL RE'S POSSESSION.

1248 — SECOND FCA IN CORE. SR32 IS WORKING FINE. NOW HAVE 2 SR CHANNELS.

1408 — Instructed Sigma Operators to check for gaps and nozzle corner offsets (EN 31007 4.5.17 a.b.+c)

1515 — Moved J02 w/554 into final loc H3 - performing new reference count rate.

1620 — Informed by SM that they will be performing charging sys test that will result in 30 gallon loss of inventory in Cavity. They will notify us before starting the test. After test, system will be secured.

1545 — Charging Sys test is complete - no impact on Reload

1545 — Swing shift on 10 FAs in Rx

K Dworski, P Racicot, Sheila S in CR; G. Sturgeon, K Connor in SFP

1713 — SIGMA can't unlatch FA J49 in Core location G-14. They will raise FA back up and re-initialize SIGMA (Note the time

1733 — FA J49 unlatched in G-14. (Yes - this was the 13th FA)

1900 — SFP called to ask about a yellow caution tag on the bridge console. Tag # 3- -99. Called OSS who said that it should have been cleared when transfer canal work was completed. Per Gerry Biggs, don't need to stop core reload, and they will resolve.

1910 — (W) crew given pre-job brief.

2005 — Changed SR31 multiplier from 10 to 100.

2025 — Received permission from SRO (Bruce Parrish) to switch between SR31+32 at our discretion - shift does not need notification each time.

* NOTE: SIGMA needs to be re-initialized often - phantom numbers on screen and index problems.

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*Note - SFF needs another headset for relieving upender operator. Tried to find another one - no success (did not check in CTMT)

2200 (W) Crew given pre-job brief

2230 SIGMA won't latch @ upender. They tried to raise the mast and re-initialize - did not work this time. Moved away and tried to reset - did not work. Bob Blumstein on his way out

2300 Mids given pre-job brief. SIGMA is tows up. At present, it is latched @ upender, but will not raise or unlatch

2315 1st C. (D. Carpenter) heading out to SIGMA. R. Van Wey notified.

2320 Mids has the watch — Jeff Camp, Mike Baldwin, Don Arey, Angelo Chase

2330 SFF crew briefs complete.

0000 FA JOB manually unlatched from SIGMA in the upender.

6/2 0030 FA JOB has been placed in cove. SIGMA was re-initialized after manual unlatching. Its working OK for now.

0110 On the next FA SIGMA lost height indication. Will pull FA back up and try again.

0115 Got the FA into cove.

0130 SIGMA blowing water out of air lines.

0245 Spoke with Dave Carpenter and Bob Blumstein. They feel that SIGMA is OK now. They bled water out of the gripper latching air line and compressor. The latch line is not continuously pressurized so it can let water leak in and the compressor moisture separator does not work well.

0300 SIGMA quit again when trying to unlatch a FA in the cove.

0315 False alarm on SIGMA, someone accidently hit the emergency stop button.

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0315 Distributed the RECO (Reasonable Expectation of Continued Operability) to the SFP upender and RE. This tells the upender operator what he is expected to verify prior to upending.

0343 SIGMA GOING TO THE UPENDER IN MANUAL because it would not go in AUTO. Once at upender it needed re-initializing again

0500 Returned SFP Bridge bypass key #59 to Shift Manager.

0600 SIGMA has been working well since about 3am with only minor exceptions.

0615 SIGMA crew was saying that the next FA move into GX would be difficult because it was between G7 and G9 which were both leaning into GX gap. They wanted permission to use SIGMA to "pull" the FA over. We used to do this in the good old days, but not today. We decided to move into GX as is and it worked. If it had not worked the only option allowed would be to use sequence deviation(s) to try to get the bucket built.

0700 SIGMA lost the mast again momentarily. Had to re-initialize

0700 DAY SHIFT ON: MARK, JAY, JOHN, CLAUDIA, BRIEFED BY JEFF CAMP.

0718 SFP Bridge checks are complete.

0720 Kathy in - relieved Jeff Camp

0803 SIGMA is acting up again. We wants to re-initialize and have Dennis Barton come look at SIGMA
FUEL MOVEMENT CONTINUES

0815 SIGMA Blowin Down Air Lines. DENNIS BARTON O.V station and checking out machine

1052 SIGMA NEEDED REINITIALIZATION

1100 SIGMA Needed Rebooting over the ~~SIGM~~ UPENDER.
1101 SIGMA REBOOTED & NO TIME - WIERD INDICATION ON SCREEN

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1227. SIGMA RE-INIALIZING NECESSARY
- SEACON ILLEGIBLE ; WOULD NOT MOVE (F6)

1230 REINTIALIZATION COMPLETE

1235 SIGMA INDICATES FULL DOWN, STILL HAS ISSUE. REQUEST USE OF BYPASS TO GO DOWN. PERMISSION FROM S.M. GRANTED

1245 SIGMA Problems. AT CORE F6. NOT SURE IF LATCHED OR UNLATCHED. TRIED to RE-INITIALIZE. HAVE A LOAD OF 400lbs Contacted DENNIS BARTON to troubleshoot

1308 OFFICALLY VERIFIED UNLATCHED AT F6 - COMING UP IN BY-PASS STILL TROUBLE-SHOOTING SIGMA. - RE-INITIALIZED

1315 SIGMA All set. GOING to UPENDER for NEXT move. DENNIS BARTON RECOMMENDS IF similar situation happens, NOT to USE BY PASS to SET DOWN. Bring F/A full up AND try again. THIS will BE Brought to RE LEAD ATTENTION

1330 NOTE TO SHIFT LEADS:
Make sure that when SF Bridge checks are performed, log in this log, tell US or SM + tell them to log in their log. Also when we have total Loss of communications, they should enter LCO.
Had discussion w/ M. Jensen + M. Reinhardt on issue of F/A being moved w/ mast at a distance of less than 2 inches while ~~trigger~~ lowe seated on pins, this is an acceptable practice as far as RE is concerned + is covered in OP 3210 B precaution limit of travel is 2" at the MOST!

1500 Swing shift in

K Dorosli, G. Sturgeon, P Racicot, K Connor, S. Stark

1530 RCS CB was 2738 @ ~1300 Shiftly crane checks done and log entry requested
SFP CB was 2803 @ 0825

1555 SM gave permission to use SIGMA bypass to disengage @ Rx E-10. After FA is unlatched, they will raise the mast and re initialize

1557 FA disengaged @ E-10

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1609 USED BYPASS TO BLOW OUT CYLINDERS ON SIGMA - WOULD NOT ENGAGE ON FA IN UPENDER.

1615 I&C WORKING ON LIMIT SWITCHES ON SIGMA - WILL BE ~ 1 HOUR THERE IS A DISCREPANCY IN POSITION INDICATION.

1650 STAND-DOWN ~ 1-2 HOURS

1745 AARON SAYS THAT PROBLEM APPEARS TO BE DUE TO CONNECTOR ON NEW CABLE - MOISTURE AND/OR NOT FULLY ENGAGED. REQUESTED EVERYONE BACK ON STATION

1815 PERMISSION GIVEN TO RECOMMENCE FUEL MOVEMENT

1900 PERMISSION TO SWITCH SR 31/32 AT OUR DISCRETION FROM S. LAWHEAD

* NOTE: RADIOGRAPHY IN CTMT 0400-0600 ∴ MUST USE SPIRAL STAIRS

* NOTE: IT IS NOW A REQUIREMENT TO HAVE UPENDER OPERATOR VERIFY THAT CART IS IN HOME POSITION (CONDITION OF OP)

1910 SRO ON SIGMA REMINDED US THAT TRI-NUC HOSES WILL NEED TO COME OUT OF THE VESSEL

2010 SFP RE REQUESTED ELEC MNTC TAKE A LOOK AT A CABLE ON THE BRIDGE WHICH APPEARS TO BE 'CHAFING'. THE ELECTRICIAN SAID THAT THERE IS NO DAMAGE, BUT IF FUEL MOVEMENT STOPS FOR A SIGNIFICANT AMOUNT OF TIME, SHOULD RECONFIGURE THE CABLE. IT'S OK TO KEEP MOVING FUEL.

2130 NEED TO COORDINATE HP TO REMOVE TRI-NUC HOSE WHEN (W) DEWATERS THE CYLINDER @ ~ MIDNIGHT (HP SAYS IT WILL ONLY TAKE THEM ~ 5 MINUTES, ONCE THEY ARE NOTIFIED.

2145 SIGMA HAD TROUBLE UNLATCHING. - GOT PERMISSION TO RAISE MAST W/ FA TO RE-INITIALIZE. IT WORKED

2205 SFP RE SAYS THAT TRANSFER CART DID NOT GET "HOME" (DID NOT ACTUATE SWITCH). THEY JOGGED IT BACK AND IT MADE IT ON THE FIRST TRY.

2245 80 FAs in the core

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2300 Midnight Shift On - Jeff Carey, Don Asay, Angelo Leale, Mike Baldwin

2325 Shiftly SFP crane checks complete, Notified SM.

6/3 0045 SIGMA is stopping temporarily to move a light and a TriNic filter hose.

0000 light and TriNic are done being moved.

0105 SIGMA IS BLEEDING LINES ON MAST AND COMPRESSOR. WHEN DONE WILL PROCEED TO UPENDER TO LATCH ON TO FA 683

0122 SIGMA HAS COMPLETED BLEEDING LINES AND IS PROCEEDING TO THE UPENDER

0130 TRANSFER CART DID NOT GET "HOME" IN SFP. CART WAS JOGGED BACK AND CART AT PIT LIGHT WENT ON AND CART WAS RAISED

0230 Delay in containment due to a stuck shoehorn.

0310 Tried to lift FA at core location R10 to get the shoe horn out. SIGMA died in doing this

0330 SM informs us that later this morning we will lose power to our shuffleworld computer for about 15 minutes, due to an UPS test.

0400 SIGMA is still broke. Dave Carpenter and Rob Blumstein are helping. They are hand cranking the FA off index and hooking height & weight to try to get the FA up into the mast.

0430 The FA is fully up into the mast. It went up on electric power but in slow speed to avoid overload. When fully up it was even a foot off on elevation.

0450 They went back to try to get the shoehorn out, but it is stuck. It did move off its initial position, rotated out, then got stuck again in a flow hole. It is a right angle shoe horn attached with aircraft cable.

0575 Our plan is to place FA TCI and HSD on the bottle with a sequence deviation. Then continue

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loading the core away from the stuck shoehorn until a recovery plan is developed.

Also, from this says that the problem with the shoehorn and the SIGMA problem were not correlated. Just double bad luck.

0545 FA J51 is unbottled a baffle location R6.

0550 Lost power to shuffleworks computer.

0613 FA H50 IS UNLATCHED AT BAFFLE LOCATION R8

0630 Distributing copies of sequence deviation to stations.

0640 SIGMA is using another shoehorn now. The elbow shoehorn is stuck. They are now using the straight shoehorn. Fuel movement is continuing

0700 Mike turned over to Days - Mark Nocera Jay Ely
John Chenis Claudia Neu

0715 Kathy in - Jeff out.

0730 Shoehorn removed from between A9/A10

0815 SFP BRIDGE CHECKS COMPLETE AND SATISFACTORY
US INFORMED.

0830 Entered sequence deviation to complete steps 182 & 184
Cannot pick up J-51 @ Romeo-6, Reprogrammed Sigma weight limits.

0925 Completed reprogramming Sigma. Headed for R-6.

0930 Latched in fla J51 in core loc R6 - headed for B10 core loc

1001 Entered Sequence deviation - complete step 250 and return to step 185

1035 RWL all set & back on track with steps 186 & 187

1148 RETRACKING MAST TO RE-INSTALL SIGMA OVER UPENDER. SIGMA IS EMPTY

1152 REFUELING SRO REPORTS A "NEAR MISS" BETWEEN SIGMA AND PERSONNEL DIRECTING MOVE WITH POLAR CRANE AUX HOOK. SM INFORMED.

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1227 Using bypass key to reinitialize SIGMA - screen full of junk - lost brains + locked up

1245 Reinitializing SIGMA over upender

1309 Reinitializing SIGMA over upender

1322 Utilizing Bypass key on Sigma (received SM permission) to enter core @ B-3

1331 Out of Bypass on Sigma, inserting on index.

1346 REINITIALIZING SIGMA OVER UPENDER.

1407 SIGMA HAVING DIFFICULTY with "HEAVY" BUNDLE IN UPENDER. Coming BACK TO FULL UP, RE-INITIALIZE & TRY AGAIN

1441 NI Ch. 32 increased X10 (14487 ct/100 sec) Momentarily

1447 F/A G61 From SFP L1 IS BEING returned to SFP RACK L1 while we try to determine what caused SPIKE on SR32.

1530 F/A 322 IS IN UPENDER, horizontal IN CONTAINMENT
All FUEL MOVEMENT IS NOW STOPPED!! UNTIL CAUSE OF SPIKE AND STATUS OF SR32 CAN BE DETERMINED. -
Checked for welding or other work - none obvious - I+C performed operability check - SAT VM performed SAT

1530 Swings crew on

Ignition 6-8-99 1515 ACOT completed on SR 32 - channel operating properly, ~~M~~ complete

1553 Permission granted by SM to resume core reload

Joe H. 6-8-99 1600 Shiftly crane checks complete/sat in SFP. US notified

1645 Reinitialized SIGMA (didn't "find bottom")

1715 Transfer cart stuck @ SFP - won't traverse to CONT, won't upend. Downpowered and reset. Still would not go to CONT. Went to home on SFP, upended, lowered. Looked like "traverse latch" was sticking, but cart is heading to CONT.

1750 SIGMA lost its brain (again); it's @ A-6 but is real sure that it's at H-6. Had good visual assurance that FA is lined up to A-6 - got permission to lower the FA. It worked.

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2100 Lost communications, apparently due to Eriksson phone network problem

1900 Late Entry - gave brief to (W) crew for Safety Standdown

2130 Eriksson person on the way in to try to fix system.

2150 Gave up on communications - Stopped fuel movement. Working on getting alternate communications setup in place.

2200 (W) crew given brief on safety stand-down

2300 119 FAs IN CORE, STILL AT A STANDSTILL; MIDS CREW IN (THANK GOD!)
Jeff Camp Dan Avery
Mike Baldwin Angelo Leone

6/4 0100 Still no communications

Eriksson - we collected up every Eriksson on the party line (plus one that wasn't). We found that the party line works except that the phone won't ring. Randy Wilson re-booted the system and it completely died.

Backup System Using Phone Lines - when using phone in CR to tie in, can only get 2 other in. Not enough. Plus the 2 field phones do not have volume control so they can barely hear the CR and can not hear each other. Plus the phone to the curtain operator is broken so he can ~~hear~~ speak but not ~~speak~~ hear.

We are now trying to connect everybody up using the Gai-Tronics intercom system. We're also working on trying to get a digital phone in CR to connect more than 2 field phones and getting volume controls to the field phones and repeater the curtain operator phone with one that works. So we may end up with a combination since the operator operations only need to talk to each other.

Shifty ITA crew leads complete.

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0200 Well the good news is that SFA RE and SIGMA are on Gai-Tronics communication with CR. Also more good news is that SIGMA and the ETS have not broken yet on midshift tonight. Bad news is that SIGMA needs more cable and splicing operations are not hooked up yet. But we are getting close.

0220 Communications re-established with Gai-Tronics. SM has given permission to start fuel movement.

0225 SIGMA got an underload going into its first core location. They are unpacking and raising the FA to fill up. Good to see that SIGMA is still working as well as it always has.

0430 SM reports that generatrix ch 2 is inop because the surveillance requires that it be able to provide audible counts to containment. This function was never installed, but in RFOT we did a temp mod to fix it. I assume we removed the temp mod without ever making the mod permanent. Generatrix ch 1 was already inop so it wasn't checked but I'm sure it would not give audible counts to containment either. The point the SM was getting at is that if we lose 31 or 32, we can't do Y/m's.

0435 SIGMA is having problems with their screen so they will raise FA and re-initialize.

0600 Fuel movement was a bit slow over on the A and B rows because they were all box moves but now that we are over open water it's going faster. SIGMA has been behaving well lately.

Shuffleboard is locked up. It lost its mind like SIGMA does.

0615 SIGMA reports they see a possible object in the next core location (K-6) so they are going to inspect it with a camera.

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6/4 0650 SIGMA reports the object is a small (~½") piece of boron. We will resume fuel movement.

0700 Day shift on - Claudia, John, Jay + Mark

0720 Kathy received turnover from Jeff

0736 SF Bridge checks complete + notified SMCUS - also confirmed audible count rates in CONC.

0955 SFP UPENDER WILL NOT GO DOWN, FA J66 CONTAINED IN UPENDER, JIM BEAUPRE IS OUT THERE INVESTIGATING.

1005 TELECOMMUNICATIONS INSTALLED IN SEPARATE LAND LINE TELEPHONE ON THE SFP RE DESK, EXT 6333.

THE OTHER PHONE (ON THE WALL) IS FOR HP & FME.

1015 CURRENT SITUATION - UPENDER HAS F/A J66 IN IT AND WON'T GO DOWN. SFP CRANE HAS H78 ON IT. H78 will be returned to M-7 in SFP.

1020 FA H78 RETURNED TO SFP M-7 FROM SFP CRANE. FA J66 STILL IN UPENDER.

1023 INITIATING SEQUENCE DEVIATION to return J66 TO SFP location A11 while checkout OF UPENDER occurs.

1100 FA J66 RETURNED TO SFP LOCATION A-11

1112 SWITCHED communications on the Gaytronics From channel 2 to channel 1. Much better. channel 2 has a faulty ground

1123 SM grants permission to use by-pass key to lower UPENDER FRAME IN SFP

1140 STAND-DOWN IN CR, SIGMA, & SRO while repairs, test are done on SFP UPENDER. UPENDER operators will stay on station.

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1145 (Late entry from 1115) TESTED COMMUNICATIONS CHANNEL 2 AND BLUE CONTROL RATES. SUCCESSFUL IN CONTROL ROOM - ANNUNCIATOR WAS TO FAINT - COULD NOT HEAR ON SIGNAL.

1240 REMANNING STATIONS - UPENDER EXERCIZED, WORKING OK, PROBLEM NOT IDENTIFIED.

1250 AUTHORIZATION RECEIVED FROM SHIFT MANAGER TO RESUME RELOAD.

1300 Resumed Reload w/ Step 273 - repeating step due to upender problem.
Got AWO from D. Beachy for using 40' pole + used pole to mechanically massage until tip lever. Also both tabs were not popping up to allow traverse. Exercised cart several times - seems to be working ok. AWO is pulled back and in possession of D Beachy / R. Manwey - YOU MUST call one of them BEFORE using pole. They will bring AWO to SFP and control the use of the poles. We MUST wait for them + they will request permission from SM BEFORE using pole on xfer cart. The poles are on the island in SFP - 20' - 2 sections.

1348 SFP UPENDER OPERATOR HEARD A RELAY CLICK ON PANEL. HE WAS BRIEFED that this COULD BE INDICATIVE OF FRAME NOT BEING ABLE TO GO DOWN. F/A WAS WITHDRAWN ABOUT 1 FT AND UPENDER WAS LOWERED SUCCESSFULLY. FRAMED UP AGAIN AND PLACED F/A IN UPENDER. RE-TAKE BLOCKS KICK UP which it is suppose to DO.

1358 UPENDER WOULD NOT FRAME-UP IN CTMT. Apparently SIGHA WAS BEING RE-INITIALIZED AND AN interlock occurred after RE-initialization, UPENDER DID FRAME-UP.

1400 Key tag #50, Key D55 is on a lanyard on the step-off pad at Spent Fuel Pool. Signed out to J.Ely. Signed in at 1500.

1500 SWINGS ON : S. SMEX, G STURGEON, P. RALICOT, K CONNOR, K BORUSKI

SFP CB : 2799 @ 1055 148 FAs IN THE CORE
RX CAV CB: 2751 @ 1025 SWINGS CREW RECEIVED SAFETY STANDDOWN BRIEF

1515 Completed safety Standdown Briefing for for all day shift crew

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1545 CART TABS DID NOT POP UP @ SFP. ADVISED UPENDER OPERATOR TO REVERSE CART AND TRY AGAIN. IT WORKED.

1552 TRANSIENT IN RAKSET II; SUSPENDED FUEL MOVEMENT WHILE OPS ASSESSES SITUATION

1600 SHIFTLY SFP BRIDGE CRANE CHECKS COMPLETE, OPS NOTIFIED

1825 Reinitialized SIGMA, normal occurrence after 7-8 moves.

1840 Cart has to be jogged twice to have tabs pop up.

2100 With SIGMA over upender and fuel assembly on hook SIGMA lost where it was. Had to be bypassed to go to full up for reinitialization. Would not let operator go to their mark for initialization. Received permission from SM (Bob Smith) to bypass SIGMA.

2110 SIGMA breakers were switched off then back on again to reinitialize and find its location.

2200 Talked with (W), advised to toggle reset switches. This worked, SIGMA reinitialized. Proceeding

2210 DUE TO INADEQUATE SPACING TO PLACE G-14 AT CORE LOCATION P-3, A SEQUENCE DEVIATION WILL BE PERFORMED PLACING G-14 AT R-5 AND REMOVE ASSEMBLY J-26 FROM N-3 TO R-7. THEN ATTEMPT TO ADJUST SPACING 1 TO 2 INCHES SOUTH. G-14 WILL ~~THEN~~ BE PLACED INTO P-3 AND THE SPACING EVALUATED AT N-3 FOR PLACEMENT OF J-26 PRIOR TO SPACING ADJUSTMENT. WESTINGHOUSE SUPERVISION AND SIGMA OPERATOR CONSULTED AND CONCURRED WITH THIS PROCESS.

2230 SEQUENCE DEVIATION WRITTEN AND DISCUSSED WITH SM - HE CONCURS AS DOES SIGMA OPERATOR. COPIES DISTRIBUTED.

2231 SEQ DEV BEING PERFORMED AS FOLLOWS: PLACE G14 FROM SIGMA INTO R5; MOVE J26 FROM N3 TO R7; "ADJUST" J53; MOVE G14 FROM R5 TO P3; MOVE J26 FROM R7 BACK TO N3.

2235 SHIFTLY CRANE CHECKS COMPLETE, SHIFT NOTIFIED.

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2300 MIDS CREW IN: Jeff Carey, Mike Baldwin, Angelo Leal, Don Arey

*NOTE - IF EVERYTHING IS COMPLETELY FINISHED DURING DAYSHIFT, PLEASE CONTACT PAUL RACICOT AND GARY STURGEON TO TELL THEM NOT TO COME IN FOR SWINGSHIFT

6/5/99 0025 Completed the sequence deviations, all fuel in the core is now in its final position, no "adjustment" of fuel was required.

Asked Roger Vansky to notify QC that their expert with two people may be required late on this shift for core map. Roger says they are at ext 4990 and ready to go.

0040 SIGMA is over the upender and not re-initialize.

0120 SIGMA reports erratic readings on their control console upender and have called Bob Blumstein. Fuel movement will continue. Also called Dave Carpenter.

0130 SIGMA has to re-initialize over one location P-4.

0150 SFP upender reports that it took several tries to get the cart to latch into position properly.

SIGMA will not grapple air hose.

0420 SIGMA is stuck over the upender. Can't go up or down.

0440 Received permission from Jim to bypass height & weight on SIGMA to re-initialize.

0500 SIGMA is unhatched from the FA and mast is raised. Height & weight are bypassed again to exercise grippers. They are working on the engage/disengage switches. Dave Carpenter and Bob Blumstein are on the scene. The disengage switch needs cleaning.

0600 WORK ON SIGMA COMPLETED. SIGMA LOWERING OVER THE UPENDER

0603 SIGMA LOADED ONTO P/A H32 IN UPENDER

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0600 Safety standdown complete for midshift REF (CR M3-99-2109).

0605 SIGMA fuel. Resuming fuel movement.

0700 DAY SHIFT ON: JEFF BRIEFED JOHN & JAY. Kathy in - had to give tour to Dave Amerine + George Pittman

0655 late entry - REINITIALIZE SIGMA

0703 REINITIALIZE SIGMA OVER UPENDER.

0735 CONFIRM CFM3 HAS AUDIBLE COUNTS CHANNEL 31.

0750 SFP BRIDGE CHECKS COMPLETE.

0908 REINITIALIZE SIGMA

0925 SPENT FUEL POOL LATCHED ON TRASH BASKET IN LOCATION U3 - WEIGHT ~ 430 lbms.

0940 TRASH BASKET GOING TO CONTAINMENT

1000 CORE RELOAD COMPLETE.

1550 Updated core map maps verified correct loading. *NOTE CORE LOCATION G12 IS IDENTIFIED AS HAVING FIA H35. THIS IS INCORRECT. REVERIFIED FIA ID IS H33 AS PER LOADING PLAN VERIFIED H35 IN CORE LOCATION B4.

6/6 0015 Performed SFP videotape mapping of fuel assembly IDs. Nearly impossible to read IDs of recently discharged G assemblies. Tapes are located in RF vertical file cabinet. Found a tie wrap lying in top nozzle of Fuel Assembly in SFP location V41.

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0120 INTERNALS LIFT SCHEDULED FOR 0130. CREW IN PLACE WILL WAIT FOR PERFORMANCE OF LOP TEST THEN PERFORM 3703D, CRDM LATCHING.

0225 HAVE PERMISSION TO SET INTERNALS.

0331 INTERNALS DOWN IN VESSEL. MAT REMOVED LIFT RIG. (W) ON WAY INTO CTMT TO PERFORM CRDM LATCHING.

0415 PERMISSION TO PERFORM 3803D - CRDM LATCHING.

0448 LATCHED DRIVE SHAFT A7.

0535 WITH HAVE OUT OF O#'S IN CORE LOCATION C9 THE TOOL IS 4 INCHES HIGHER.

0555 STILL DON'T KNOW WHAT IS WRONG WITH C9. WILL MARK AND PROCEDURE ~~OF MODIFY~~ WILL COME BACK TO C9 LATER.

0600 C7 IS 3-4" HIGHER. HAVE CONFIRMED POSITIVE ENGAGEMENT DUE TO WEIGHT VERIFICATION UNLATCHING TOOL. DRIVING SIGMA NORTH LOADING STATION. (W) WANTS TO GO TO PITTSBURG.

0620 Communications turned over to David Peffer in control room

0621 Latched on C-5

0622 C-5 engaged correctly and lowered to normal height

0625 Latched on D-2; engaged correctly and lowed to normal height.

0630 Latched on D-4; engaged correctly and lowered to normal height

0635 Latched on D-8; engaged did not lower to normal height

0655 Latched on D-12; engaged did not lower to normal height

0700 Latched on D-14; engaged and lowered to normal height

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0730 ALL OFF STATION - EVALUATING WHAT TO DO NEXT. INVESTIGATING HOW TO MEASURE DRAG FORCES.

0830 ALL HANDS ON STATION, SM INFORM TO WAIT UNTIL COMPLETION OF COP POST. ~ 30-45 M.N.

1136 Shift manager gave approval to commence latching CRDMs

1140 Latched on E13 and Engaged and lowered to normal height

1143 Latched on E11 engaged and lowered to normal height

1145 Latched on E5 engaged and lowered to normal Height

1148 Latched on E3 engaged and lowered to normal height

1151 Latched on F2 engaged and lowered to normal height

1152 Latched on F6 engaged and lowered to normal height

1153 Latched on F8 engaged and lowered to normal height

1155 Latched on F10 engaged and lowered to normal height

1157 Latched on F14 engaged and lowered to normal height

1159 Latched on G15 engaged and lowered to normal height

1202 Latched G13 engaged and lowered to normal height

1205 Latched G3 engaged and lowered to normal height

1207 Latched H2 engaged and lowered to normal height

1208 Latched H4 engaged and lowered to normal height

1210 Latched H6 engaged and lowered to normal height

1212 Latched H8 engaged and lowered to normal height

1214 Latched H10 engaged and lowered to normal height

1215 Latched H12 engaged and lowered to normal height
Lifted again weight normal, lowered to normal

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1218 Latched H14 engaged and lowered to normal height

1220 Latched J13 engaged and lowered to normal height

1220 Latched J3 engaged and lowered to normal height

1224 Latched J1 engaged and lowered to normal height

1228 Latched K2 engaged and lowered to normal height

1230 Latched K6 engaged and lowered to normal height

1231 Latched K8 engaged and lowered to normal height

1233 Latched K10 engaged and lowered to normal height.

1235 Latched K14 engaged and lowered to normal height.

1237 Latched L13 engaged and lowered to normal height

12-39 Latched L11 engaged and lowered to normal height

1241 Latched L5 engaged and lowered to normal height

12:44 Latched L3 engaged and lowered to normal height

12:47 Latched M2 engaged and lowered to normal height

12:49 Latched M4 engaged and lowered to normal height

12:51 Latched M8 engaged and lowered to normal height

12:53 Latched M12 engaged and lowered to normal height

12:55 Latched M14 engaged and lowered to normal height

12:54 Latched N11 engaged and lowered to normal height

13:01 Latched N9 engaged and lowered to normal height

13:03 Latched N7 engaged and lowered to normal height

13:04 Latched N5 engaged and lowered to normal height

13:07 Latched P4 engaged and lowered to normal height

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1425 E5 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1427 E3 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1431 C7 ENGAGED, DRAG-TEST SAT, RETURED TO NORMAL HEIGHT

1435 C9 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1438 D08 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1440 D12 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1444 D14 ENGAGED, DRAG-TEST SAT, RETURN TO NORMA HEIGHT

1447 F02 ENGAGED, DRAG-TEST SAT, RETURN TO NORMAL HEIGHT.

1449 F06 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1451 F08 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1454 F10 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1456 F14 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1459 G15 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1501 G13 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1504 G03 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1507 H2 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1509 H4 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

512 H6 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

14 H8 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

17 H10 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

1521 H12 ENGAGED, DRAG-TEST SAT, RETURNED TO NORMAL HEIGHT

523 H14 ENGAGED, DRAG-TEST SAT. RETURNED TO NORMAL HEIGHT

525 J13 ENGAGED, DRAG-TEST SAT, RETURND TO NORMAL HEIGHT

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1425 E5 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1427 E3 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1431 C7 ENGAGED, DRAG TEST SAT, RETURED TO NORMAL HEIGHT

1435 C9 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1438 D08 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1440 D12 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1444 D14 ENGAGED, DRAG TEST SAT, RETURN TO NORMA HEIGHT

1447 F02 ENGAGED, DRAG TEST SAT, RETURN TO NORMAL HEIGHT.

1449 F06 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1451 F08 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1454 F10 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1456 F14 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1459 G15 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1501 G13 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1504 G03 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1507 H2 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1509 H4 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1512 H6 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1514 H8 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1517 H10 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1519 H12 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1523 H14 ENGAGED, DRAG TEST SAT. RETURNED TO NORMAL HEIGHT

1525 J13 ENGAGED, DRAG TEST SAT, RETURND TO NORMAL HEIGHT

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1529 J03 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1530 J01 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1533 K2 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1536 K6 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1538 K8 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1541 K10 ENGAGED, DRAG TEST SAT, RETURN TO NORMAL HEIGHT

1543 K14 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1546 L13 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1548 L11 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1551 L5 ENGAGED, DRAG TESTED SAT, RETESTED DUE TO OBSERVING SOMETING ABNORMAL, NOTHING FOUND ON REDCAE, RETURNED TO NORMAL HEIGHT.

1556 L3 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1559 M2 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1602 M4 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

1605 M8 ENGAGED, DRAG TEST SAT, RETURNED TO NORMAL HEIGHT

16:09 M12 Engaged, Drag Test Sat, Returned to Normal Height

16:12 M14 Engaged, Drag Test Sat, Returned to Normal Height

16:14 N11 Engaged, Drag Test Sat, Returned to Normal Height

16:16 N9 Engaged, Drag Test Sat, Returned to Normal Height

16:19 N7 Engaged, Drag Test Sat, Returned to Normal Height

16:21 N5 Engaged, Drag Test Sat, Returned to Normal Height

16:24 P4 Engaged, Drag Test Sat, Returned to Normal Height

16:26 P6 Engaged, Drag Test Sat, Returned to Normal Height

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16:28 P8 Engaged, Drag Test Set, Returned to Normal Height

16:30 P10 Engaged, Drag Test Set, Returned to Normal Height

16:33 P12 Engaged Drag Test Set, Returned to Normal Height

16:35 R9 Engaged, Drag Test Set, Returned to Normal Height

Core Alts complete - SRO removed communications w/ control room removed.

6/8/99
0130 Verified control rod drive shafts were latched by observing as cavity level drained to where the buttons all were uncovered at the same time. Buttons were uncovered when level on LT 462 (cold calibrated) reached approximately 17.8%. First time, we got down to 20.4% and OPS started filling the cavity by performing SIH testing. Second time was successful. There is a video of this, but don't know if you can tell much, as I did not get VP to zoom in their camera enough.
Kent U.

1/8/99 Removed the following Gaicsson Rings to Communications

6105	6975
6972	6633
6968	6961
6942	
6976	

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