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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400 -LA
(Shearon Harris Nuclear	)	ASLBP No. 99-762-02-LA
Power Plant)	)	
	)	

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)

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January 4, 2000

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I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.113, Orange County hereby submits a detailed written summary and sworn submission (hereinafter "Summary") of all the facts, data, and arguments which are known to the County and on which the County proposes to rely at the January 21, 2000, oral argument. This Summary presents Orange County's legal and factual grounds for asserting that Carolina Power & Light's ("CP&L's") application to amend its Operating License by expanding the capacity of spent fuel pool storage pools at the Harris nuclear power plant fails to satisfy the criticality prevention requirements of General Design Criterion ("GDC") 62 and applicable NRC guidance, and fails to provide adequate protection of public health and safety to members of the public living in the vicinity of the Harris plant.[1]

As required by 10 C.F.R. § 2.1113(b), the factual assertions in this Summary are submitted under the sworn declaration of Dr. Gordon Thompson, the County's expert witness regarding criticality prevention issues. A further declaration of Dr. Thompson's qualifications and experience and a description of his work on this Summary is attached as Exhibit 1.

As detailed below, this summary demonstrates that as a matter of law, CP&L'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. If the Board does not find that the issue can be disposed of clearly as a matter of law, the County submits that it has submitted substantial evidence that there is a genuine and substantial factual dispute between CP&L and the County regarding whether the criticality prevention measures it has elected are acceptable under GDC 62 and applicable portions of the NRC Staff's regulatory guidance, and whether there is any basis for finding that the public health and safety can be adequately protected by CP&L's proposed criticality prevention measures.

II. STATEMENT OF THE CASE

This case 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

1 *See* Letter from James Scarola, CP&L, to NRC, re: Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment, Spent Fuel Storage

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 NRC Staff's various guidance documents related to criticality prevention do contain some provisions which are consistent with GDC 62 and which provide assistance in determining whether the physical criticality prevention measures that are designed to prevent criticality in normal operation will also suffice to protect public health and safety under accident conditions. In order to evaluate the effectiveness of criticality prevention in an accident, it is necessary to perform a criticality analysis that encompasses possible accident scenarios and evaluates the efficacy of criticality prevention measures during each scenario. A useful tool for such an analysis is the Double Contingency Principle, which is set forth in Draft Regulatory Guide 1.13, a document employed by the Staff for evaluating criticality analyses. This version of the Double Contingency Principle requires that a criticality analysis must demonstrate that criticality could <u>not</u> occur without at least two unlikely, independent and concurrent failures or operating limit violations. In order to make a meaningful application of the Double Contingency Principle, it is necessary to identify what are possible sets of unlikely, independent and concurrent failures or operating limit violations, and then evaluate those events in combination to determine whether the facility's criticality prevention arrangements will maintain subcriticality during each set of events. Draft Reg. Guide 1.13 also advises that in evaluating such accident scenarios, it may be assumed that initial conditions are in the normal range. However, the deterioration of those conditions in the course of each accident scenario must also be examined. In this case, CP&L has neither complied with GDC 62, nor has it made a reasonable application of the Double

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(December 23, 1998), (hereinafter "License Amendment Application).

Contingency Principle. CP&L proposes to rely for criticality prevention on the control of burnup/enrichment levels, which necessarily entails ongoing administrative procedures and controls. These procedures are not only prohibited by GDC 62, but they are inherently less reliable than physical systems and processes. CP&L has also misapplied the Double Contingency Principle, by failing to identify and evaluate the sets of unlikely, independent, and concurrent failures that could lead to a criticality accident. Instead, CP&L has addressed only one scenario in which criticality is approached: the mispositioning of a single fresh PWR fuel assembly in pool C or D.

Because it has made no attempt to identify and evaluate the sets of events that could lead to a criticality accident, CP&L has no basis for asserting that its analysis of a single event is conservative. Moreover, experience at operating nuclear power plants shows that a single error can lead to the mispositioning of multiple fuel assemblies, and that mispositioning of this kind is a likely event. Given the potential for mispositioning of multiple assemblies, CP&L's and the NRC Staff's own criticality calculations show that the spent fuel in pools C and D could become supercritical.

Accordingly, because it violates GDC 62 and misapplies the valid portions of applicable NRC Staff guidance for the conduct of criticality accident analyses, CP&L's License Amendment Application must be rejected.

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 accident 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 measures for criticality prevention.[2] 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

2 For a more complete discussion of the language and history of GDC 62, see Section IV.

IV.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."[3] 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").[4] 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 utilize available storage space."[5] 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.6

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below.

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

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

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

6 *Id.*

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")[7]. 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 "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.[8] Draft Reg. Guide 1.13 also calls for the application of the Double Contingency Principle, articulating the principle as follows:

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

8 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.

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

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

A more recent guidance document on criticality prevention in spent fuel storage pools is a Memorandum from Laurence Kopp, NRC, to Timothy Collins, NRC, re: Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants (August 19, 1998) (hereinafter "Kopp Memorandum").[9] 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:

> 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.[10]

The Kopp Memorandum thus effectively reduces the double contingency principle to a "single

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

contingency principle."[11]

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 measures and reducing the rigor of the accident analysis required.

3. Evolution of Criticality Prevention in Fuel Pools

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

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10 *Id.*, Attachment 4.

11 A more detailed discussion of the Kopp Memorandum appears in Appendix A to this

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.[12]

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

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Summary.

12 *See* US 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

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.[13] Most, perhaps all, fuel pools at US nuclear plants have been

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Light Water Power Reactor Fuel, NUREG-0404 (2 volumes) Appendices B and D (March 1978).

13 In pursuit of even higher storage densities in fuel pools, the nuclear industry has also studied fuel storage options involving a reduced presence of water between the fuel rods. Water moderates fast neutrons, so a reduced presence of water can yield a subcritical margin of reactivity even as the spacing between fuel assemblies or rods is reduced. One water-displacing option is to place spent fuel assemblies inside cans and to fill all empty space inside each can with small metal beads, thereby achieving a fuel storage density of 0.75 tonne U per square foot. A second option is to compact fuel assemblies by crushing the fuel spacers until rods are nearly touching, thus achieving a fuel storage density of about 0.95 tonne U per square foot. A third option is to dismantle the fuel assemblies and store the rods in close contact with each other inside cans, thus achieving a fuel storage density of about 1.1 tonne U per square foot. None of these options has been generally adopted. *See* US Department of Energy, Spent Fuel Storage Fact Book, DOE/NE-0005 (April 1980).

equipped for some years with racks that incorporate solid boron within the rack structures, often in the form of Boral panels.

d. Ongoing administrative measures

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 measures 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 methods, the density of storage of intact fuel assemblies in a 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.[14] 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.

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

e. Independent Spent Fuel Storage Installations

There is an alternative to adopting ever-higher densities of fuel storage in an existing fuel pool. That alternative is to construct an independent spent fuel storage installation ("ISFSI"). ISFSI's have been built at several US nuclear plant sites. In each case, a dry storage technology has been employed. As of September 1998, installations of this kind were licensed at 11 nuclear plant sites.[15]

B. The Harris License Amendment Application

There are four spent fuel storage pools at Carolina Power & Light Company's ("CP&L's") Harris nuclear power plant. Only two of the pools, designated "A" and "B," are currently in operation. At present, pool A contains 6 PWR racks with a total of 360 spaces, and 3 BWR racks with a total of 363 spaces. Pool B contains 12 PWR racks with a total of 768 spaces and 17 BWR racks with a total of 2,057 spaces. Under the present license, one additional BWR rack with a total of 121 spaces could be placed in pool B.

CP&L now seeks a license amendment to activate pools "C" and "D."[16] The purpose of the license amendment is to allow CP&L to use the Harris facility to store spent fuel generated at CP&L's one-unit Harris PWR station, its two-unit Brunswick BWR station, and its one-unit Robinson PWR station. If granted, the license amendment would allow the placement in pool C of up to 11 PWR racks with a total of 927 spaces and 19 BWR racks with a total of 2,763 spaces;

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

16 CP&L's proposed changes to its Technical Specifications are described in Enclosure 5 to the License Amendment Application. Enclosure 7 is a non-proprietary report entitled "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools 'C' and 'D' (Rev. 2). By letter dated March 17, 1999, CP&L submitted Rev. 3 to Enclosure 7, which reflects the release of some information that previously had been considered proprietary. Aside from the additional disclosures, the content of Rev. 3 is the same as Rev. 2.

and the placement in pool D of 12 PWR racks with a total of 1,025 spaces. CP&L envisions this placement occurring in three campaigns in pool C, followed by two campaigns in pool D.

For all four spent fuel pools at Harris, CP&L intends to ensure that Keffective will be less than or equal to 0.95 when the racks are flooded with unborated water, including an allowance for uncertainties[17] The proposed means for achieving this objective for pools C and D are different from the means for preventing criticality in pools A and B, however. For pools A and B, a subcritical margin of reactivity is now achieved during normal operation in two ways: through the rack's neutron-absorbing properties, which are enhanced by incorporating solid boron in the rack structures; and by maintaining a nominal 10.5 inch center-center distance in the PWR racks and a nominal 6.25 inch center-center distance in the BWR racks. These conditions will continue to apply in pools A and B after pools C and D are activated.

For pools C and D, CP&L proposes to space the PWR spent fuel assemblies significantly closer together than they are placed in pools A and B. A nominal 9.017 inch center-center distance will be maintained in the PWR racks, while a nominal 6.25 inch center-center distance will be maintained in the BWR racks. The PWR rack spacing is close to the smallest distance that is physically possible for intact PWR fuel, because the PWR fuel assemblies used in the Harris reactor have a square cross-section that is 8.43 inches wide.[18] For this configuration, the distance between the fuel assemblies and the neutron-absorbing properties of the racks, taken together, will not be sufficient to maintain the desired subcritical margin of reactivity under normal conditions, still less under accident conditions. Therefore, CP&L proposes to introduce an additional means of criticality suppression for PWR fuel in pools C and D.

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17 Keffective is the neutron multiplication factor in a finite array of fuel, allowing for neutron leakage.

CP&L proposes to introduce new, ongoing administrative measures that would limit the combination of burnup and enrichment of the PWR spent fuel in pools C and D to an "acceptable range." The range of acceptable burnup/enrichment values is shown in Figure 5.6.1 of the proposed technical specifications, Enclosure 5 to the License Amendment Application. According to CP&L: "The burnup criteria will be implemented by appropriate administrative procedures to ensure verified burnup as specified in the proposed Regulatory Guide 1.13, Revision 2, prior to fuel transfer into Spent Fuel Pools C or D."[19] CP&L further states that: "Strict administrative controls will prevent an unacceptable assembly, as determined by the acceptance criteria stated in Section 4.2, from being transferred to Harris Pools C and D."[20]

According to CP&L, burnup is not a criterion of acceptability for storage of BWR fuel in pools C and D. The reactivity of an acceptable BWR fuel assembly will be limited by restricting its U-235 enrichment to 4.6 wt% and by the requirement that, for a Standard Cold Core Geometry ("SCCG") array of the fuel, Kinfinite must be less than or equal to 1.32 at all times during the life cycle of the assembly.[21] CP&L calculations indicate that a BWR assembly of the type to be placed in pools C and D will, in a SCCG array, be maximally reactive (i.e., exhibit its

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18 *See* Harris FSAR Table 1.3.1-1, Amendment No. 30.

19 License Amendment Application, Enclosure 7, Revision 3 at 4-4.

20 *Id.* at 4-17. CP&L's License Amendment Application does not provide details about these administrative controls. In its June 14, 1999, RAI Response (Exhibit 5), CP&L provides some information about the controls that will apply to PWR fuel from the Robinson station. *See* Exhibit 5. However, that information is not sufficient to support a thorough assessment of CP&L's administrative controls, including an assessment of their probability of failure. Similarly, none of the documents provided by CP&L during the discovery phase of this proceeding provide sufficient information about relevant administrative controls to support an assessment of their efficacy. In a deposition, CP&L employee provided general information about CP&L's computer program for tracking the movement of fuel at the Harris plant, but was unfamiliar with the details of the program, such as how information used in the program is verified. *See* Transcript of Deposition Michael J. DeVoe, P.E. at 9-25 (October 20, 1999), attached as Exhibit 6.

maximum value of Kinfinite) when its burnup is approximately 12,000 MW-days per tonne U.22

C. Orange County's Intervention in Licensing Proceeding

On January 7, 1999, the NRC published a notice of opportunity for a hearing on the proposed license amendment, at 64 Fed. Reg. 2,237. Orange County filed a timely hearing request and intervention petition on February 12, 1999. On April 5, 1999, Orange County submitted contentions challenging the adequacy of the License Amendment Application. Orange County's contentions included a challenge to the adequacy of CP&L's criticality measures. The claims raised by the contention were two-fold. First, Orange County contended that CP&L's proposed reliance on Draft Regulatory Guide 1.13, which permits reliance on administrative measures for criticality prevention, was precluded by GDC 62, a duly promulgated regulation. GDC 62 requires the use of "physical systems and processes." Second, Orange County argued that even if CP&L could rely on the regulatory guidance, it could not satisfy the "double contingency" principle set forth in the Draft Reg. Guide:

> At all locations in the reactor 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 concurring failures or operating limit violations.

Draft Reg. Guide 1.13 at 1.13-12 (emphasis in original). CP&L's proposed administrative controls on criticality would not satisfy this requirement because only one failure or violation, namely placement in the racks of PWR fuel not within the "acceptable range" of burnup, could cause criticality. Orange County's Supplemental Petition to Intervene at 10-13.

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21 Kinfinite is the neturon multiplication factor in an infinite array of fuel.
22 *See* page 4-10 of Revision 3 of Enclosure 7 to license amendment application. *See also* letter from Donna B. Alexander, CP&L, to U.S. NRC, enclosing response to April 29, 199, Request for Additional Information (June 14, 1999) (hereinafter "June 14, 1999 RAI Response"), attached as Exhibit 5.

In LBP-99-25, Memorandum and Order (Ruling on Standing and Contentions), the Licensing Board ruled that Orange County had standing, and admitted two of the County's contentions. 50 NRC 25 (1999). As admitted by the Licensing Board, Contention TC-2 (Inadequate Criticality Prevention) reads as follows:

> CONTENTION: Storage of pressurized water reactor ("PWR") spent fuel in pools C and D at the Harris plant, in the manner proposed in CP&L's license amendment application, would violate Criterion 62 of 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 GDC 62, CP&L proposes to prevent criticality of PWR fuel in pools C and D by employing administrative measures which limit the combination of burnup and enrichment for PWR fuel assemblies that are placed in those pools. This proposed reliance on administrative measures rather than physical systems or processes is inconsistent with GDC 62.

50 NRC at 35. In ruling on the contention, the Licensing Board used CP&L's "two-basis construct," construing the bases of the contention as follows:

a. Basis 1 -- CP&L's proposed use of credit for burnup to prevent criticality in pools C and D is unlawful because GDC 62 prohibits the use of administrative measures, and the use of credit for burnup is an administrative measure.

b. Basis 2 -- The use of credit for burnup is proscribed because Regulatory Guide 1.13 requires that criticality not occur without two independent failures, and one failure, misplacement of a fuel assembly, could cause criticality if credit for burnup is used.

The Board found that that the first basis raises "essentially a question of law," and that the second basis raises the following "question of fact":

> Will a single fuel assembly misplacement, involving a fuel element of the wrong burnup or enrichment, cause criticality in the fuel pool, or would more than one such misplacement or a misplacement coupled with some other error be needed to cause such criticality?

LBP-99-25, 50 NRC at 36.[23]

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23 As discussed below in Section IV.H and in Appendix A, the Board's summary of the

As required by 10 C.F.R. § 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 argument is held. CP&L invoked the hybrid hearing process, and therefore this Summary is being filed herewith.

ARGUMENT

IV. THE PROPOSED LICENSE AMENDMENT FAILS TO COMPLY WITH GDC 62 BECAUSE IT IMPROPERLY RELIES ON ADMINISTRATIVE MEASURES FOR CRITICALITY PREVENTION.

As demonstrated below, the proposed License Amendment Application fails to comply with GDC 62 because it improperly relies on administrative measures 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. Orange County 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 CP&L's proposed criticality prevention measures. If the Board decides that it is unable to rule for Orange County on these submissions, the Board should find that Orange County has raised a genuine, substantial and material factual and legal dispute with CP&L, and order that Contention TC-2 proceed to a trial pursuant to 10 C.F.R. § 2.1115.

Double Contingency Principle as found in Draft Reg. Guide 1.13 is not fully consistent with the language of the Reg. Guide itself, or with Orange County's contention. Orange County does not believe, however, that the Board intended to issue a definitive interpretation of the Draft Reg. Guide with this admissibility ruling.

As discussed in more detail in Section I of Orange County's Detailed Summary and Sworn Submission of Facts, Data and Arguments, etc., With Respect to Quality Assurance Issues, 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).[24]

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

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24 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 the County'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, Orange County has provided the Board with evidence of (a) the clear basis for distinguishing physical measures from ongoing administrative measures, 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 constitutes a physical measure for purposes of eliminating a boiling, misting or

physical systems or processes. The phrase "physical systems or processes" is not 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 measures. Standing alone, the plain language of GDC 62 clearly dictates that CP&L must rely solely on physical measures to avoid criticality. Because CP&L intends to rely in part on ongoing administrative measures, *i.e.*, control of burn-up 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 measures. 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 measures, on the other hand.

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foam environment in a spent fuel pool has effectively been mooted.

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 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).[25]

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

25 A copy of this Information Notice is attached to Appendix A as Exhibit A-16.

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.

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 Orange County's knowledge, a set of draft General Design Criteria first appeared as an attachment to an Atomic Energy Commission ("AEC")[26] press release of November 22, 1965, entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits."[27] 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

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

27 The Press Release and attached documents are attached as Exhibit 7.

> 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 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.[28]

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.[29]

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.[30] 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".[31] The set of

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28 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 8.

29 The October 25, 1966, letter and attached draft are attached to this Summary as Exhibit 9.

30 The February 8, 1967 letter and attached draft are attached to this Summary as Exhibit 10.

31 Note by the Secretary, W.B. McCool, to AEC Commissioners re: Proposed Amendment to

GDCs was described as a proposed amendment to 10 CFR 50. The potential 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).[32] 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).[33] 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 as geometrically safe

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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 11.

32 A copy of the Federal Register notice is attached to this Summary as Exhibit 12.
33 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 13.

configurations shall be used to insure that criticality cannot occur.'[34]

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.[35] 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.[36] 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

On February 20, 1971, the AEC published the General Design Criteria in final form.37 The introduction to the GDC's now characterized them as "minimum requirements" for the design of

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34 *Id.*, Attachment containing "Specific Comments" at 11.

35 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 14.

36 *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 15.

37 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 16.

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.

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.[38] 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."[39] 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 features to prevent inadvertent criticality. The inadvertent criticality that 10 CFR 70.24 is intended to address could only occur during fuel-handling operations.
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> 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

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38 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.
39 62 Fed. Reg. at 63,825, Col. 3.

> 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.
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> 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.*[40]

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.[41] 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

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40 62 Fed. Reg. at 63,825-26. (emphasis added)

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

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.
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> (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 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

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1998).

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.[42] 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.

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 measures:

> 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 1.0) without considering the presence of soluble boron in the

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42 As discussed in note 23 above, arrangements for storage of fresh fuel in a pool should also

water.[43]

It should also be noted that the type of ongoing administrative measure proposed by CP&L in the instant case, *i.e.*, control of burnup/enrichment levels in the fuel, 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 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 plant operating licenses. It is not being considered under Part 72, the ISFSI regulations.

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

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ensure that the fuel remains subcritical in the presence of boiling water, foam or mist.

43 The other provisions of § 50.68, subsections (b)(5) through (8), are not relevant to this proceeding.

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.[44] 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

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44 *See* examples cited in Appendix B: Braidwood Unit 1, (July 10, 1996); Cooper Station (March 5, 1990); Crystal River Unit 3 (November 9, 1987); Oyster Creek Unit 1 (January 21,

promulgated in 1988, when the Commission added requirements for Monitored Retrievable Storage ("MRS") to the ISFSI regulations.[45] 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 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.

1987).

45 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).

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.[46]

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.

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46 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.

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.[47]

E. The Administrative Criticality Prevention Proposed by CP&L Would Violate GDC 62.

As described above in Section III.B, CP&L proposes to restrict the burnup/enrichment of PWR fuel in order to suppress criticality under normal conditions. CP&L asserts that these burnup/enrichment limits will be carried out through "strict administrative controls" that will prevent an unacceptable assembly from being transferred to Harris Pools C and D.[48]

This reliance on ongoing administrative procedures and controls to enforce

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47 *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).

burnup/enrichment limits 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, CP&L 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.[49]

F. CP&L'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 TC-2, CP&L and the NRC Staff argued that its reliance on control of burnup/enrichment levels 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 the Board 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

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48 License Amendment Application, Enclosure 7 Rev. 3 at 4-17.

49 In one criticality analysis, CP&L relied on the presence of soluble boron during an accident. *See* CP&L's June 14, 1999, RAI Response (Exhibit 5). In a subsequent response to the same RAI, CP&L stated that a new criticality analysis shows that if defined as Kinfinite less than 1, subcriticality can be maintained in unborated water, in the presence of one mispositioned fresh PWR fuel assembly. Letter from Donna B. Alexander to U.S. NRC (October 15, 1999), attached as Exhibit 17. However, a soluble boron concentration of 400 ppm was found necessary to "maintain Kinfinite less than the regultory limit of 0.95." *Id.* As discussed below in Section IV.F, the consideration of mispositioning of a single fresh fuel assembly does not constitute an adequate criticality analysis. For this reason, and to meet the regulatory limit of 0.95 for Kinfinite, it is necessary consider whether CP&L's reliance on the presence of soluble boron under abnormal conditions is consistent with GDC 62.

regulation is controlling. A regulation has the force of law; in comparison, a regulatory guide is a set 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 CP&L Nor the Staff Has Demonstrated That Public Health And Safety Will Be Adequately Protected If CP&L Relies on Ongoing Administrative Measures for Criticality Control.

Although the Staff's regulatory guidance is fundamentally at odds with GDC 62, the Staff's practice of permitting ongoing administrative measures for the prevention of criticality in spent fuel pools is well-entrenched. In recent years, the NRC Staff has approved many applications similar to CP&L'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.[50] 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 discussed in Appendix B, the limited information that was provided by the Staff in discovery, and that Orange County was able to find in the Public Document Room, 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

50 Orange County 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

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 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, including Harris pools C and D. 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 the Harris nuclear power plant.

H. CP&L's Criticality Accident Analysis Misapplies Applicable Staff Guidance.

As discussed above, CP&L's criticality analysis is fundamentally deficient because CP&L relies on administrative measures 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 CP&L'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

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record of relevant events. *See* Appendix C.

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." CP&L has misapplied this guidance in four principal respects. First, CP&L ignores the words "at least," and evaluates only one failure instead of sets of 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; and fourth, it unreasonably assumes that a single error can lead to the mispositioning of only one fuel assembly.

Before addressing CP&L's misapplication of the Draft Reg. Guide in more detail, it is necessary to point out that in admitting "Basis 2" of Contention TC-2, the Board summarized the thrust of the contention in a manner that is overly narrow and inconsistent with the contention.[51] The Board's summary of Basis 2 shortens Draft Reg. Guide 1.13's statement of the Double

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51 The Board characterized Basis 2 as follows:

> Basis 2 – The use of credit for burnup is proscribed because Regulatory Guide 1.13 requires that criticality not occur without two independent failures, and one failure, misplacement of a fuel assembly, could cause criticality if credit for burnup is used.

The Board also found that:

> The second basis raises a question of fact: Will a single fuel assembly misplacement, involving a fuel element of the wrong burnup or enrichment, cause criticality in the fuel pool, or would more than one such misplacement or a misplacement coupled with some other error be needed to cause such criticality?

LBP-99-25, 50 NRC at 36.

Contingency Principle from "at least two independent, unlikely, and concurrent failures" to "two independent failures." The decision also contains language implying the assumption that one failure would lead to the misplacement of no more than one fuel assembly, and that the Double Contingency Principle is a single failure criterion. The Board also refers to "the required single failure criterion," when in reality the criterion is a double contingency standard.

Orange County believes that in admitting Basis 2 of Contention TC-2, the Board intended to permit the litigation of whether CP&L's criticality analysis satisfies the accident analysis criteria set forth in Draft Reg. Guide 1.13, as quoted and discussed by by Orange County at page 12-13 of its Supplemental Petition to Intervene.[52] Orange County does not interpret the Board's summary of the contention's basis to constitute a definitive interpretation of Draft Reg. Guide 1.13, which after all is the subject of the contention. As the Board noted in admitting Basis 2, "Clearly the nature of this amendment, introducing as it does the presence of high density racks

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52 The contention stated as follows:

> Draft Reg. Guide 1.13 does not support the administrative measures proposed by CP&L. Although Appendix A contains some language implying that the design of spent fuel racks against criticality can take credit for burnup (pages 1.13-13, 14, 15), other parts of the Draft Reg. Guide clearly proscribe such activity. For instance, at page 1.13-9, the Draft Reg. Guide states that:
>
>> At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurring failures or operating limit violations.
>
> (emphasis in original). CP&L's proposed administrative controls on criticality would not satisfy this requirement because only one failure or violation, namely placement in the racks of PWR fuel not within the "acceptable range" of burnup, could cause criticality. Note that "misplacement of a spent fuel assembly" is identified in the Draft Reg. Guide as one of nine "credible normal and abnormal operating occurrences."

The contention did not summarize Draft Reg. Guide 1.13 or assert that Orange County's only on the site, involves a change that may call into question conformance with this aspect of the regulations." *Id.* at 36. In order to evaluate whether the License Amendment Application complies with this provision of Draft Reg. Guide 1.13, it is necessary to closely examine each aspect of the Double Contingency Principle as set forth in the Draft Reg. Guide, without attributing the Board's general summary of the Draft Reg. Guide as a definitive interpretation of its meaning.

CP&L's criticality accident analysis for pools C and D violates the guidance of Draft Reg. Guide 1.13 in the following respects:

1. CP&L ignores the words "at least," and evaluates only one failure instead of sets of failures.

Draft Reg. Guide 1.13 calls for the analysis of situations involving "at least" two failures or violations of operating limits. Analysis that meets this requirement must identify the sets of failures or violations that might cause criticality, and then evaluate these failures or violations in combinations of at least two, to determine which combinations will cause criticality. This process will yield an "envelop" of criticality which bounds the combinations of failures and violations that produce criticality. That envelope cannot be identified if failures or violations are evaluated one at a time. When the envelope has been identified, the Double Contingency Principle can be applied, with consideration as to whether failures or violations are unlikely, independent and concurrent. *See* Appendix C for a more detailed discussion.

CP&L has not gone through this process, but has only considered a single failure, limited to the mispositioning of one fresh PWR fuel assembly.

2. CP&L fails to determine what failures are "unlikely, independent, and concurrent."

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concern was the misplacement of a single fuel assembly.

When the envelope of criticality has been determined for a particular situation, such as the storage of PWR fuel in Harris pools C and D, application of the Double Contingency Principle requires a determination, for each failure or violation represented in the envelope, as to whether that failure or violation is unlikely, and whether it is independent of and concurrent with the other failures or violations represented in the envelope. For Harris pools C and D, the most significant failures or violations will be fuel mispositioning events and boron dilution events. CP&L has failed to determine if these events are unlikely, independent, or concurrent.

3. CP&L assumes that mispositioning of fuel is an "unlikely" event when in fact it is likely.

In considering possible criticality accidents at Harris pools C and D, CP&L assumes that the mispositioning of fuel is an unlikely event. CP&L offers no evidence to support this assumption. In fact, as shown in Appendix B and discussed in Appendix C, experience shows that fuel mispositioning is likely. Moreover, in a criticality accident involving fuel mispositioning and soluble boron dilution, these events will typically be consecutive rather than concurrent. High-reactivity fuel could be mispositioned in a fuel pool prior to or after a boron dilution event, or at both times if an event sequence involving mispositioning of multiple fuel assemblies spans a time period during which boron dilution occurs. Were CP&L to treat fuel mispositioning as a likely occurrence, then the criticality analysis would necessarily consider fuel mispositioning in combination with a complete absence of soluble boron, even employing the invalid, non-conservative version of the double Contingency Principle which is articulated in the Kopp Memorandum. Similarly, were CP&L to consider mispositioning and soluble boron dilution as consecutive occurrences, the criticality analysis would necessarily consider these occurrences in combination. Calculations by CP&L and the NRC Staff, summarized in

Appendix C, show that mispositioning of a single fresh PWR fuel assembly in Harris pools C or D would, in the absence of soluble boron, cause Keffective to exceed the regulatory limit of 0.95. Mispositioning of more than one assembly could result in a supercritical configuration, potentially critical on prompt neutrons alone.

4. CP&L unreasonably assumes that a single error can lead to the mispositioning of only one fuel assembly.

In considering the role of fuel mispositioning as a potential cause of criticality, CP&L has restricted its attention to the mispositioning of only one PWR fuel assembly. Underlying this restriction is an assumption that a single failure or violation will lead to the mispostioning of only one fuel assembly. In fact, as demonstrated in Appendix B and discussed in Appendix C, experience shows that a single error can lead to the mispositioning of multiple fuel assemblies.

In addition to its improper reliance on administrative measures for criticality control, CP&L's misapplication of the Double Contingency Principle in the manner discussed above has yielded a criticality analysis that is non-conservative and inadequate to provide a reasonable assurance that public health and safety will be protected in the event of an accident. Whether or not the administrative measures chosen by CP&L are approved by the Licensing Board as consistent with GDC, CP&L's methodology for performing its criticality accident analysis must be rejected as inconsistent with valid and applicable NRC Staff guidance.

V. CONCLUSION

For the foregoing reasons, the criticality prevention measures proposed in CP&L's License Amendment Application for the expansion of spent fuel storage capacity at Harris must be rejected as inconsistent with GDC 62 and valid and applicable NRC Staff guidance. Moreover, CP&L's criticality prevention measures are demonstrably insufficient to provide a reasonable level of protection to public health and safety.

Orange County has demonstrated that the License Amendment Application must be rejected as a matter of law. If the Board declines to reject the application as a matter of law, it should find that Orange County has raised material and substantial issues of law and fact, and order the parties to proceed to an adjudicatory hearing on Contention TC-2.

Respectfully submitted,

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I, Dr. Gordon Thompson, declare under penalty of perjury that the technical facts presented in the above Summary and Sworn Submission, including its appendices, are true and correct to the best of my knowledge and that all expressions of opinion regarding technical matters are based on my best professional judgment.

G. R. Thompson

Gordon Thompson, Ph.D.

January 4, 2000

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, including mispositioning of fuel assemblies in spent fuel storage racks, other fuel management errors, a soluble boron dilution event, other errors in managing soluble boron, and erroneous criticality calculations. These incidents shed light on the potential for inadvertent criticality in fuel pools.

The original source of information on the incidents described here was a set of Licensee Event Reports (LERs) supplied to Orange County by the NRC Staff during discovery in the operating license amendment proceeding regarding CP&L's proposal to increase spent fuel storage capacity at the Harris nuclear power plant.

The historical record summarized here is almost certainly incomplete, for three reasons. First, the LERs supplied by the NRC Staff were not systematically selected through a search of the full body of LERs, and the NRC Staff does not keep a database of incidents relevant to mispositioning of fuel or the dilution of soluble boron. Second, each relevant incident that has been identified by a nuclear plant licensee was not necessarily reported to the NRC by submission of an LER. 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 descriptions are arranged by alphabetic order of the plants where the incidents occurred.

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

On August 21, 1996, an analysis of blackness test[2] data was received by the licensee, indicating shrinkage and gaps in the Boraflex in the spent fuel racks.

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

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.

Braidwood Unit 1: July 10, 1996 (Licensee Event Report 456/96-008-00 (August 5, 1996))[3]

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.

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[2] 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.

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

Braidwood Unit 1: June 17, 1996 (Licensee Event Report 456/96-007-00 (July 15, 1996))[4]

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 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))[5]

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.

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

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

Byron Station: May 28, 1996 (Licensee Event Report 454/96-008-00 (June 25, 1996))[6]

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.

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))[7]

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.

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

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

The reviewing engineer's failure to detect both of the NMC's mistakes arose from the reviewing engineer's use of inappropriate procedures.

Catawba Unit 1: March 5, 1990 (Licensee Event Report 413/90-016-00 (April 19, 1990))[8]

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.

Cooper Station: November 18, 1986 (Licensee Event Report 298/86-034-00 (December 18, 1986))[9]

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

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

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

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))[10]

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.

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 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.

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

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

McGuire Unit 1: July 11, 1994 (Licensee Event Report 369/94-005-00 (August 10, 1994))[12]

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 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."

McGuire Unit 1: October 24, 1991 (Licensee Event Report 369/91-016-00 (November 25, 1991))[13]

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.

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

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

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.

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

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.

Oconee Unit 1: January 8, 1996 (Licensee Event Report 269/96-001-00 (February 7, 1996))[15]

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 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;

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

15 A copy of this LER is attached as Exhibit A-14.

(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))[16]

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 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.

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

Susquehanna Unit 1: October 6, 1993 (NRC Information Notice 94-13, (February 22, 1994))[17]

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.

Three Mile Island Unit 1: February 4, 1998 (Licensee Event Report 289/98-002-01 (April 3, 1998))[18]

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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17 A copy of this Information Notice is attached as Exhibit A-16.

18 A copy of this LER is attached as Exhibit A-17.

Waterford Station: February 18, 1994 (NRC Information Notice 94-13, Supplement 1 (June 28, 1994))[19]

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

Various plants and incidents (NRC Information Notice 94-13 (February 22, 1994))[20]

Various fuel-handling incidents occurred at Vermont Yankee, Peach Bottom, Susquehanna and Nine Mile Point 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."

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[19] A copy of this Supplement is attached as Exhibit A-18.
[20] See Exhibit A-16.

EXHIBIT B-1

Braidwood Unit 1:
LER 456/96-010-02 (August 11, 1998)

(4-95)

LICENSEE EVENT REPORT (LER)

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Braidwood Unit 1	05000456	1 of 8

TITLE (4) Failure to Comply With Design Basis Due to Degradation of Boraflex in Spent Fuel Racks and a criticality analysis modeling deficiency

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	21	96	96	010	02	08	11	98	Braidwood Unit 2	05000457
									Byron Unit 1 and 2	05000454 & 05000455

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(iii)	☐ 73.71(b)
☐ 20.2203(a)(1)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(iv)	☐ 73.71(c)
☐ 20.2203(a)(2)(i)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(v)	☐ OTHER
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(viii)(A)	(Specify in Abstract
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)	☐ 50.73(a)(2)(viii)(B)	below and in Text,
☐ 20.2203(a)(2)(v)	X 50.73(a)(2)(ii)	☐ 50.73(a)(2)(x)	NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. Schliessmann, Licensing Engineer	(815) 458-2801 Extension 2038

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	DB	Boraflex Panel	B959	N					

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines 16)

Analysis of Neutron Attenuation test data for Braidwood's Spent Fuel Racks received on 8/21/96, shows Boraflex shrinkage and gaps. The largest gap has a width of greater than four inches. A gap of greater than four inches in any Boraflex panel exceeds that assumed in the current criticality analysis. The spent fuel storage racks are designed to maintain a Keff ≤ 0.95 when flooded with unborated water. The cause of this event was determined to be failure of the Boraflex due to deterioration as a result of improper material selection. Corrective actions include controls on Spent Fuel Pool (SFP) boron concentration and silica concentration. The safety analysis contained in this report concludes that there is reasonable assurance that the Braidwood SFP will maintain a Keff ≤ 0.95.

On 3/25/97, a modeling deficiency was identified in criticality analysis CAC-96-248, "Byron and Braidwood Spent Fuel Rack Criticality Analysis with Credit for Soluble Boron", dated October 31, 1996. This analysis assumed Boral poison plates were located on <u>all</u> four faces of all Region 1 storage cells. The criticality model did not reflect the actual (as designed) configuration of the Boral poison plates, which are located on the interior portions of the new Region 1 fuel storage racks but are not present on the periphery of the Region 1 storage cells. Subsequent to the discovery of this modeling deficiency, supplemental criticality analyses for the actual Region 1 cell Boral geometries were

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit(s): 1	Event Date: 08/21/96	Event Time: 1224 Hours
Reactor Mode(s): 1	Power Level(s): 100%	RCS [AB] Temp./Press. NOT / NOP

SUPPLEMENTAL INFORMATION:

Unit(s): 1	Event Date: 03/25/97	Event Time: 1700 Hours
Reactor Mode(s): 1	Power Level(s): 100%	RCS [AB] Temp./Press. NOT / NOP

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event that contributed to the severity of the event.

On 8/21/96, analysis results of Neutron Attenuation (Blackness) test data were received at Braidwood Station, indicating shrinkage and gaps in the Boraflex in the spent fuel racks. The largest gap has a width greater than four inches. A gap of greater than four inches in any Boraflex panel exceeds that assumed in the current criticality analysis. An ENS phone call was made at 1349.

The Spent Fuel Pool (SFP) at Braidwood Station has fuel racks installed that utilize sheets of Boraflex for reactivity suppression. Boraflex is constructed of an organic polymer with a silica filler and neutron absorbing boron carbide interspersed within the silica filler.

In 1987, ComEd first identified gamma-radiation induced damage to the Boraflex polymer. The damage progresses through two stages. First, the Boraflex cracks and shrinks, producing cracks and gaps. The second phase occurs after the polymer has sustained significant damage, and consists of the Boraflex becoming brittle and susceptible to dissolution in the Spent Fuel Pool cooling water.

The reactivity effects associated with the first stage have been characterized in the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Considering Boraflex Gaps and Shrinkage," Westinghouse, June 1994, supplemental criticality analysis. Sufficient margin exists within this supplemental criticality analysis to accommodate the anticipated levels of cracking and gapping associated with the first stage of degradation.

The second stage of damage involves long-term degradation of the Boraflex. The second stage appears to commence after the Boraflex has received approximately 4E9 RADs of gamma exposure. There are a number of variables (burnup, cooling time, recent power history, etc.) that affect the exposure rate. The presence of silica in the SFP cooling water is another indicator that storage locations have progressed into the second stage of damage. The reason for the uncertainty in the rack's condition lies in the degradation mechanism associated with the second stage. The second stage involves slow dissolution of the Boraflex. The rate of dissolution is determined by the concentration of reactive silica in SFP solution, thermally-induced flow velocities, and coolant temperature inside of the storage racks. The larger the panel spacing, the stronger the local flow and thus the dissolution rate increases.

The recent Blackness Testing campaigns at Byron and Braidwood indicate progress into the second stage of damage has occurred, and that the maximum gap width allowed in the current criticality analysis has been exceeded.

Based on the above facts, Braidwood already has large numbers of storage locations in the second stage of degradation. The degradation mechanism associated with the second stage proceeds slowly, however it is both difficult to predict and measure the extent of damage. Although Blackness Testing is useful for measuring cracks, gaps, and wastage, it does not measure an overall reduction in boron density. Therefore Blackness Testing provides incomplete information regarding the current state of a given storage location. An approved methodology to measure boron spatial density does not currently exist for PWRs. Therefore, the gaps recently found at Braidwood Station may not represent the full extent of Boraflex degradation.

When assessing the current state of the storage racks, the following factors, along with others, are considered: they include the slow nature of the degradation process, the continued presence of some Boraflex, the successful performance of the surveillance coupon program, the inclusion of Boral in the Region 1 rack design, and the potential for additional reactivity margins due to burn-up.

ComEd has performed calculations to support a short term recommendation of maintaining greater than 2000 PPM soluble boron in the Spent Fuel Pool to compensate for the degradation of the Boraflex. These calculations are very conservative. The 2000 PPM limit is intended to approximate the total reactivity suppression worth of the installed Boraflex in both the Region 1 and Region 2 fuel storage racks. Therefore, even if all Boraflex were to be removed from the Spent Fuel Racks, the 2000 PPM value is adequate to maintain the Spent Fuel Pool at ≤ 0.95 Keff.

Based on the recent Blackness Test data, it cannot be stated with certainty that Technical Specification 5.6.1.1 is met. This specification states, "The spent fuel storage racks are designed and shall be maintained with a Keff ≤ 0.95 when flooded with unborated water,...". Therefore, the racks are in an "Indeterminate" state of operability as defined in NRC Generic Letter 91-18, they have been conservatively declared inoperable, and compensatory measures that were initiated in 1995 were verified.

This event is being reported pursuant to 10CFR50.73(a)(2)(ii)(B) - any event or condition that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant

On 03/25/97, additional reviews by ComEd identified a modeling deficiency in criticality analysis CAC-96-248, "Byron and Braidwood Spent Fuel Rack Criticality Analysis with Credit for Soluble Boron", dated October 31, 1996. This analysis was performed to support Technical Specification Amendment No. 86 for Byron Units 1 & 2 and Amendment No. 78 for Braidwood Units 1 and 2, issued April 2, 1997. This "interim" criticality analysis was performed due to the degradation of the Boraflex in the spent fuel racks. The deficiency is due to inadequate modeling of the physical configuration of the Boral panels within the Byron and Braidwood Region 1 Fuel Storage Racks.

Due to ComEd's concerns regarding the industry's experiences with Boraflex degradation during the mid-80's, Boral panels were placed in the flux traps of the Region 1 racks during initial fabrication. The Boral panels were inserted into the flux traps that exist between each cell within a Region 1 rack. Criticality analysis CAC-96-248 included in the assumptions of the model for the Region 1 racks

- The array is infinite in lateral (x and y) extent. There is no interface requirements between Region 1 storage racks.
- Boral poison plates were on all four faces of all storage cells.

The criticality model did not reflect the actual (as designed) configuration of the Boral plates, which are located on the interior portions of the new Region 1 fuel storage racks, but were not designed to be installed on the periphery of the Region 1 storage cells. Thus, Region 1 periphery storage cells actually contain Boral plates on only three sides; and the four corner cells actually contain only two interior Boral plates.

In March 1997, the Westinghouse criticality engineer was reviewing the SFP rack peripheral geometry in an attempt to regain storage locations that were lost due to constraints required in the 1996 analysis. Drawings of the Byron and Braidwood SFP and racks were supplied to Westinghouse at their request. On March 20, the Westinghouse criticality engineer contacted ComEd Nuclear Fuel Services (NFS) with a concern that there may not be Boral on the peripheral walls of the Region 1 racks due to the girdle bar geometry. As a result of subsequent ComEd reviews of rack drawings and discussions with the vendors responsible for construction and seismic analysis of the racks, ComEd concluded that Boral poison plates were neither present on the periphery of the Region 1 storage cells nor designed to be in these locations.

C. CAUSE OF EVENT:

The cause of this event was determined to be failure of the Boraflex due to deterioration as a result of improper material selection.

In 1987, ComEd first identified gamma radiation-induced damage to the Boraflex polymer. That damage progresses through two stages. First, the Boraflex cracks and shrinks, producing cracks and gaps. Second, after the polymer has sustained significant damage, the Boraflex becomes brittle and is susceptible to dissolution in the Spent Fuel Pool cooling water.

The cause of the 3/25/97 event was determined to be the result of a modeling error in vendor performed criticality analyses, and inadequate reviews of the analyses input and assumptions against manufacturing drawings during the criticality analyses reviews and verifications. The infinite array criticality analysis methodology was not appropriate for the unique placement of Boral in the Region 1 racks, and did not properly model the interface between Region 1 racks.

Boral panels were placed in the flux traps of the Region 1 racks during initial fabrication. The manufacturing drawings for the Region 1 racks are inadequate to determine Boral panel placement, and no as-built drawings of the Region 1 racks were ever generated. The original criticality analysis for the Region 1 racks was performed in 1987 by a vendor. The criticality "model", originally generated by the vendor, was transferred to Westinghouse, and was carried forward through all subsequent criticality analyses. The original analysis and all subsequent analyses assumed for Region 1 an infinite array and Boral poison plates on all four faces of all cells. Criticality analysis CAC-96-240 assumed that the Boraflex poison was removed from the storage racks, and that the Boraflex was replaced with water. The analysis also modeled the Region 1 storage cells with Boral panels on all four faces. This and previous analyses did not specifically model the peripheral Region 1 cells that do not have Boral panels on their exterior faces.

D. ASSESSMENT OF SAFETY CONSEQUENCES:

Recent Blackness Testing indicates that the degradation of the Braidwood Spent Fuel Racks exceeds that assumed in the criticality analysis. This could lead to a condition where the Technical Specification reactivity limits for the SFP could be exceeded.

Based on a comparison with prior analyses by ComEd Nuclear Fuel Services for the Byron/Braidwood reactor cores, maintaining SFP boron concentration >2000 PPM will ensure that the requirements for maximum reactivity in the SFP are met, even assuming the Boraflex panels are ineffective from a reactivity mitigation standpoint. The analysis assumed enriched fuel with no burnup (e.g. maximum reactivity) in close proximity to other assemblies. The physical separation of assemblies in the Spent Fuel Racks is greater than the separation in the core. In addition, the spent fuel assemblies are at much lower reactivity due to burnup from incore operation. For these reasons, there is reasonable assurance that the Braidwood Spent Fuel Pool maintains a Keff ≤ 0.95.

After discovery of the modeling deficiency on 3/25/97, supplemental criticality analyses for the actual Region 1 cell Boral geometries were performed and demonstrated that, with administrative controls in place regarding boron concentration and fuel placement in Region 1 rack interface, acceptance criteria for spent fuel storage is met. The supplemental criticality analyses utilized the same assumptions, codes, procedures, and uncertainties used to support the 1996 criticality analysis (CAC-96-248) but with Boral panels located only in the interior cell-to-cell interfaces. The supplemental analyses modeled the following Region 1 rack geometries:

1. Corner cell of rack facing two concrete walls.
2. Peripheral cell of rack facing one concrete wall.
3. Empty row of cells facing a full row of cells across a Region 1 to Region 1 rack interface.
4. Checkerboard pattern of cells across a Region 1 to Region 1 rack interface.

Calculations were performed for the four rack geometries to verify that with a maximum nominal enrichment of U-235, that Keff is less than 1.0. The analyses ignored the presence of Boraflex and accurately modeled Boral only on the interior rack faces. This calculation was performed with no soluble boron assumed present in the SFP. The resulting reactivities were compared to the all cell Keff calculated in section 3.2.1 of CAC-96-248. The all cell Keff (from CAC-96-248) was verified to be greater than the reactivities calculated for these four rack geometries. The biases and uncertainties calculated in CAC-96-248 remain valid for use with the four analyzed rack geometries. By determining that the all cell Keff remains bounding, the conclusions of CAC-96-248 are applicable for the four analyzed rack geometries analyzed for the following acceptance criteria:

1. Assuming no soluble boron, the maximum nominal enrichment of U-235 could be stored and a Keff of less than 1.0 is maintained,
2. Taking credit for a minimum concentration of soluble boron of 2000 ppm, a Keff of less than or equal to 0.95 is maintained, and
3. Assuming the SFP water temperature postulated accident and taking credit for a minimum concentration of soluble boron of 2000 ppm, a Keff of less than or equal to 0.95 is maintained.

Additional cases for the misloaded assembly were performed. These cases were calculated at no soluble boron conditions and the resulting reactivities were shown to be less than the all cell Keff from CAC-96-248. The dropped assembly accidents are not affected by the Boral configuration.

An additional criticality analysis was performed taking credit for 2000 ppm soluble boron and no Boral and no Boraflex present in the spent fuel racks. This analysis verified that Keff was less than or equal to 0.95 for all storage locations based on fuel assembly locations at the time of the event discovery.

The supplemental criticality analyses are conservative since, in reality, an appreciable amount of Boraflex remains in place in addition to the administrative requirement to maintain at least 2000 ppm in the SFP. It is concluded that the safety analysis impact due to the incorrect modeling of the Boral configuration is minimal.

E. <u>CORRECTIVE ACTIONS:</u>

The following are actions being taken to either minimize the Boraflex degradation or mitigate the effects of Boraflex degradation.

Evaluation has shown that 2000 PPM soluble Boron will compensate for even fully deteriorated Boraflex. Therefore, Braidwood will administratively maintain >2000 PPM soluble Boron until further review of the Boraflex issue. This will be tracked by NTS item #456-180-96-01001.
This item has been completed.

Spent Fuel Pool silica reduction using Reverse Osmosis will be restricted until the licensing amendment to allow for soluble boron credit is approved. This will be tracked by NTS item #456-180-96-01002.
This item has been completed.

The long term corrective action for this situation consists of submittal of a licensing amendment to allow soluble boron to be credited in maintaining the pool ≤ 0.95 Keff. The analysis for this amendment is in progress. Submittal to the NRC is expected in mid-1997. This will be tracked by NTS item #456-180-96-01003.
This item has been completed.

ComEd has created a Boraflex Issue Committee to work with the industry to resolve this issue.
Resolution: The revised Braidwood Criticality Analysis does not credit Boraflex. This item has been completed.

An effectiveness review will be performed for all corrective actions listed above. This will be tracked by NTS item #456-180-96-010ER.
This item has been completed.

As a result of the supplemental criticality analyses for the actual Region 1 cell Boral geometries, the following administrative control has been implemented in station procedures (BwAP 2364-9, paragraph C.1). This change is tracked by NTS item # 456-180-96-010S101. "No assembly may be placed in a Region 1 rack location face adjacent to another assembly across a Region 1 rack interface."
This item has been completed.

Based on this additional administrative control, Braidwood station repositioned fuel assemblies in the spent fuel pool. ComEd has subsequently verified that the storage configuration of fuel assemblies in Byron and Braidwood SFP meet the criteria specified in CAC-96-248 and meet the supplemental criticality analyses performed for the actual Region 1 cell Boral geometries.
This item has been completed.

ComEd will review the spent fuel pool criticality analysis for other ComEd facilities that may be susceptible to similar problems. These reviews will verify that the current analyses conservatively consider the potentially limiting geometries associated with peripheral cells of adjacent fuel racks, especially as it relates to the placement of fixed poisons such as Boral or Boraflex on the outer faces or peripheral cells. This will be tracked by NTS item #456-180-96-0105102.
This item has been completed.

NFS will submit required reading for the entire staff to clarify the responsibilities of NFS engineers when performing an "acceptance review" of externally generated calculations: (1) verify all ComEd specific inputs to the analysis (such as physical dimensions, setpoints, and limits), and (2) verify that the vendor's methodologies and assumptions are valid when applied to ComEd. This will be tracked by NTS item #456-180-96-0105103.
This item has been completed.

NFS will revise NFS procedures governing the review and approval of controlled work to clarify the responsibilities of NFS engineers when performing an "acceptance review" of externally generated calculations: (1) verify all ComEd specific inputs to the analysis (such as physical dimensions, setpoints, and limits), and (2) verify that the vendor's methodologies and assumptions are valid when applied to ComEd. This will be tracked by NTS item #456-180-96-0105104.
This item has been completed.

A review of regulatory requirements/guidance on fuel pool rack criticality analysis will be performed to ensure other requirements are adequately addressed. This will be tracked by NTS item #456-180-96-0105105.
This item has been completed.

Obtain As-built drawings for the fuel pool racks. This will be tracked by NTS item #456-180-96-0105106.
Delete corrective action:
Communications with the vendor confirmed as-built drawings were not generated, however, the station does maintain the design drawings for the fuel pool racks. The intent of the corrective action was to ensure future critically analyses correctly model the lack of boral poison plates on the periphery of the racks. The revised criticality analyses and procedures support the current configuration.

The Boraflex and Criticality Analysis issues have been submitted to the ComEd Part 21 Committee for consideration of reportability under 10CFR Part 21. This review will be tracked by NTS item # 456-180-96-0105107.
This item has been completed.

An effectiveness review will be performed for all corrective actions initiated as a result of the 3/25/97 event. This will be tracked by NTS item #456-180-96-0105IER.
This item has been completed.

NRC FORM 366A (4-95)
U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND THE PAPERWORK REDUCTION PROJECT

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Braidwood Unit 1	05000456	96	010	02	8 of 8

(If more space is required, use additional copies of NRC Form 366A)(17)

F. **PREVIOUS OCCURRENCES:**

Boraflex degradation, that was bounded by the current Criticality Analysis, was previously identified at Braidwood Station (NTS # 456-201-95-2155). Anticipatory compensatory and mitigating actions were put in place and included: administratively maintaining the SFP boron concentration greater than 2000 ppm, maintaining the SFP temperature as low as possible, restricting the removal of Silica from the SFP, minimizing transfer of SFP water into the RCS during refueling operations, and developing a Boraflex committee to review and approve long term solutions to the Boraflex degradation problem for ComEd. The corrective actions of the previously identified problem would not have prevented continuing deterioration of the Boraflex.

Prior activities were reviewed to determine if precursor events occurred or if prior activities may have prevented the event. The absence of Boral poison plates on the periphery cells could have been identified during the blackness testing in 1991. The increase in neutron signal may have been attributed to degradation of Boraflex rather than lack of Boral poison plates. The interpretation of test results may have been skewed by the belief that Boral poison sheets were on the periphery cells (now known not to be accurate).

A review of industry events did not find previous occurrences of errors in criticality analyses due to Boral poison sheets not being installed.

G. **COMPONENT FAILURE DATA:**

MANUFACTURER	NOMENCLATURE	MODEL	MFG. PART NO.
Bisco Products Inc.	Boraflex Panels	NA	NA

EXHIBIT B-2

Braidwood Unit 1:
LER 456/96-008-00 (August 5, 1996)

NRC FORM 366 (4-95) — U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Braidwood Unit 1	05000456	1 OF 6

TITLE (4)
Improper Placement of Spent Fuel Resulting in Technical Specification Violation Due to Personnel Error.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
07	10	96	96	– 008 –	00	08	05	96	FACILITY NAME	DOCKET NUMBER

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)								
POWER LEVEL (10)		20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
		20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
		20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
		20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. Lawson, System Engineering	(815) 458-2801 x3061

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
NA									

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

During the verification of Spent Fuel Pool storage locations, it was discovered that one fuel assembly was stored in Region 2, and not in the required checkerboard configuration, based upon the burnup versus initial enrichment limits specified by Tech. Spec. 5.6.1.1.b.2. The cause of this event was personnel error. The burnup versus initial enrichment limits, which determine acceptable fuel storage configurations, were changed by Technical Specification Amendment 58. A calculation performed prior to this change to verify that the new limits were met contained an incorrect burnup. The calculation was not independently verified, so the error was not identified. Immediate corrective actions were to relocate Assembly S46W into Region 1 of the Spent Fuel Pool. Additional corrective actions were counseling of the individual regarding expectations and procedure revision. This event resulted in no safety concerns. Two previous fuel mispositioning events were due to personnel error and procedural and management deficiencies.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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Braidwood Unit 1	05000456	96	--008--	00	2 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

A. PLANT CONDITIONS PRIOR TO EVENT:

UNIT: Braidwood Unit 1 EVENT DATE: 07/10/96
EVENT TIME: 1045
MODE: 1 RX POWER: 100
RCS [AB] TEMPERATURE/PRESSURE: NOT/NOP

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event that contributed to the severity of the event.

On May 28, 1996, nuclear engineers at Byron Station reported that fuel assemblies were mislocated in Region 2 of the Spent Fuel Pool that did not meet the requirements of Technical Specification 5.6.1.1.b.2, "Fuel Storage - Region 2". This situation had resulted from a change in Spent Fuel Pool storage requirements, caused by Amendment 58 to the Technical Specifications, approved on January 20, 1995. On 7/10/96, as a part of the investigation into this event, ComEd Nuclear Fuel Services transmitted a listing of fuel not meeting the burnup versus initial enrichment limitations to Braidwood Station. Braidwood Station personnel immediately noted that the Nuclear Fuel Services transmittal identified 84 assemblies that should be either located in Region 1 or in a checkerboard configuration, but only 83 assemblies were stored to meet this requirement. Upon verifying the information, Braidwood personnel identified that fuel assembly S46W was improperly loaded into a close-packed configuration in Region 2 of the Spent Fuel Pool without meeting the burnup versus initial enrichment requirements of Technical Specification 5.6.1.1.b.2. Upon discovery, fuel assembly S46W was immediately relocated to Region 1.

Fuel Assembly S46W was discharged from the reactor core during A1R02 on October 11, 1991. In accordance with normal Braidwood Station practices, it was originally placed into Region 1 of the Spent Fuel Pool. S46W was relocated into Region 2 of the Spent Fuel Pool on June 16, 1992. Prior to

B. DESCRIPTION OF EVENT (continued)

this move, procedure BwAP 2364-9, "Controlling Movements of Nuclear Fuel Into The Spent Fuel Racks", was performed to verify that all moved assemblies met the burnup-initial enrichment criteria. At the time of the move, the Technical Specifications were met for assembly S46W, based on assembly burnup, supplied by Nuclear Fuel Services.

Technical Specification Amendment 58 was incorporated on January 20, 1995, to reflect a new criticality analysis that includes fuel enrichment to 5.0 weight percent uranium 235, and to incorporate a 3 percent uncertainty to account for inaccuracies in calculation of assembly burnup.

The Nuclear Material Custodian at that time performed calculations, using the new limits, before moving fuel into Region 2 of the Spent Fuel Pool during A2R04 (refueling outage prior to Unit 2 Cycle 5). Although these calculations were not required until receipt of the approved Amendment, they were performed to verify that the new limits would be complied with upon approval. These calculations were performed during October of 1994.

The Nuclear Material Custodian also performed calculations on all fuel assemblies in the Spent Fuel Pool at that time to check whether the previously discharged fuel assemblies met the new criteria. He performed this calculation using a spreadsheet program, which was not independently verified. This spreadsheet was later transmitted to Nuclear Fuel Services as part of the investigation into the Byron event. The spreadsheet calculation failed to identify that assembly S46W did not meet the new limits because the fuel assembly burnup as provided by Nuclear Fuel Services was incorrectly entered.

BwAP 2364-9, "Controlling Movements Of Nuclear Fuel Into The Spent Fuel Racks", Revision 1, does not require an independent review of calculations, is not retained as plant documentation, and requires performance only upon movement within the Spent Fuel Pool. Since independent verification is not required, the Nuclear Material Custodian was misled into thinking that independent verification was not required for the calculations prior to Amendment incorporation. Since performance is not required except prior to fuel movement in the Spent Fuel Pool, calculations were not required prior to amendment incorporation, when the burnup versus initial enrichment limits changed.

This event is being reported pursuant to 10CFR50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Braidwood Unit 1	05000456	96	--008--	00	4 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

C. CAUSE OF EVENT:

The cause of this event was personnel error.

The Nuclear Material Custodian at the time of incorporation of Technical Specification Amendment 58 should have performed calculations to verify compliance with the new limits as a Controlled Analysis, with independent verification and retention for the duration of the Operating License for Braidwood Station.

D. SAFETY ANALYSIS:

There were no safety consequences from this event. The Spent Fuel Pool boron concentration remained well above the value assumed for the current criticality analysis, while all fuel assemblies adjacent or near to the misloaded assembly had burnups higher than the burnup assumed in the criticality analysis. If the Spent Fuel Pool boron concentration had been at the value assumed for the current criticality analysis, no safety consequences would have occurred because the amount of fissile fuel contained within the mispositioned fuel assembly was bounded by the existing analysis, and all adjacent fuel assemblies had burnup greater than the minimum burnup assumed. If the Spent Fuel Pool boron concentration had been at the value assumed for the current criticality analysis and an additional fuel misloading had occurred, the required k-eff of 0.95 may have been exceeded.

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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Braidwood Unit 1	05000456	96	--008--	00	5 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

E. CORRECTIVE ACTIONS:

The Nuclear Material Custodian at the time of incorporation of Technical Specification Amendment 58 has been counseled regarding failure to meet expectations.

Procedure BwAP 2364-9, "Controlling Movements Of Nuclear Fuel Into The Spent Fuel Pool", will be revised to require independent verification of the calculations, retention as plant documentation, and performance when the burnup versus initial enrichment limits are changed. This will be tracked to completion by NTS item #456-180-96-00801.

The location of all fuel assemblies in the Spent Fuel Pool will be verified by direct observation using an underwater camera. This will be completed prior to moving any fuel presently located in the Spent Fuel Pool, unless such movement is required to ensure safety. This will be tracked to completion by NTS item # 456-180-96-00802.

A review of the effectiveness of corrective actions taken for this event will be conducted by one year following completion. This will be tracked to completion by NTS item # 456-180-96-00803.

F. PREVIOUS OCCURRENCES:

LER 1-96-007 involved failure to comply with Technical Specification 5.6.1.1 due to positioning fuel that did meet the burnup versus initial enrichment limits in a close-packed configuration immediately adjacent to fuel that did not meet the limits in a checkerboard configuration. The causes of this event were personnel error and procedural and management deficiencies. Although the LER 1-96-007 event resulted in mispositioning of nuclear fuel within Region 2 of the Spent Fuel Pool, the circumstances leading to this event were different from those leading to the subject event.

Additionally, one other occurrence involving fuel mispositioning (457-200-94-016) was noted. A review of the event determined that new fuel was mispositioned in the Spent Fuel Pool during transfer from the New Fuel Storage Vault. The cause of the event was personnel error due to a lack of a questioning attitude and failure to follow procedures. A review of the corrective actions determined that they would not have prevented this event from occurring.

NRC FORM 366A (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Braidwood Unit 1	05000456	96	--008--	00	6 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL	MFG PART NO.
N/A			

EXHIBIT B-3

Braidwood Unit 1:
LER 96-007-00 (July 15, 1996)

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Braidwood Unit 1	05000456	1 OF 6

TITLE (4)

Improper Placement of Spent Fuel in Regards to Checkerboarding Due to Personnel Error, and Procedural and Management Deficiencies.

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	17	96	96	– 007 –	00	07	15	96	FACILITY NAME	DOCKET NUMBER

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	01		20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)
POWER LEVEL (10)	100		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)

	50.73(a)(2)(viii)
	50.73(a)(2)(x)
	73.71
	OTHER
	Specify in Abstract below or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
D. Lawson, System Engineering	(815) 458-2801 x3061

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
NA									

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 6/17/96, fuel was repositioned in the Spent Fuel Pool into a configuration that was not bounded by the existing Criticality Analysis. The mispositioning was identified by the Fuel Handling Supervisor accomplishing the fuel movement and reported to the Nuclear Material Custodian (NMC). The fuel in the inappropriate configuration was immediately repositioned. Investigation concluded that the NMC and Independent Reviewers did not consider the effects on lower burnup fuel in adjacent storage locations in planning the fuel moves. Additionally, this configuration restriction had not been properly transmitted to Braidwood Station by the analysis vendor. Causes of the event were determined to be personnel error, and procedural and management deficiencies. Corrective actions taken involve preparation of a new procedure containing more detailed position guidance, counseling of personnel involved, revising the NMC qualification guide, reviewing requirements for other fuel stored in the Spent Fuel Pool, and immediate repositioning of the fuel to an appropriate configuration. A safety analysis determined that the mispositioned fuel did not cause a criticality concern. A previous event involving fuel mispositioning was caused by a failure to follow procedures for fuel moves

NRC FORM 366A
(5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Braidwood Unit 1	05000456	96	--007--	00	2 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

A. PLANT CONDITIONS PRIOR TO EVENT:

UNIT: Braidwood Unit 1 EVENT DATE: 6/17/96
EVENT TIME: 1212
MODE: 1 RX POWER: 100
RCS [AB] TEMPERATURE/PRESSURE: NOT/NOP

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event that contributed to the severity of the event.

Fuel moves were planned for the Spent Fuel Pool in preparation for "Blackness Testing". "Blackness Testing" consists of a technique in which a neutron source is used to evaluate the degradation of the Boraflex neutron absorber material in the Spent Fuel Pool storage racks. Continued periodic testing is a commitment to the NRC to ensure that neutron moderation remains within acceptable bounds, ensuring that the Spent Fuel Pool Criticality Analysis assumptions remain valid. Nuclear Component Transfer Lists (NCTLs) were prepared by the Nuclear Material Custodian (NMC) for this purpose on 5/9/96. The NMC preparing these moves had recently assumed the NMC position. An independent review of the NCTLs was performed by the previous NMC, on 5/11/96. As a part of the normal review process, a second independent review was conducted by the Station Reactor Engineer (SRE), on 5/15/96. On 6/17/96 at approximately 0930, during the performance of these fuel moves, the Fuel Handling Supervisor noted a fuel configuration that he considered to be suspect. The suspect configuration involved irradiated fuel stored in Region 2 of the Spent Fuel Pool. Requirements for storage of fuel in Region 2 are that either the fuel must have a specified burnup corresponding to its initial enrichment, or it must be stored in a "checkerboard" configuration if its initial enrichment was less than or equal to 4.2 weight percent Uranium 235. Fuel meeting the burnup-initial enrichment restriction may be stored in any configuration in Region 2. The suspect fuel configuration involved fuel that met the burnup-initial enrichment restriction being stored in a close-packed configuration immediately adjacent to fuel that did not meet the requirement, and was placed into the "checkerboard"

B. DESCRIPTION OF EVENT (continued)

configuration. The Fuel Handling Supervisor immediately contacted the System Engineer in charge of the "Blackness Testing", who then contacted the NMC. After consulting with the SRE, the NMC directed the Fuel Handling Supervisor to suspend fuel movement, and began preparing NCTL Variations (BwAP 370-3T3) to reposition the suspect fuel assemblies pending further investigation. The NCTL Variations were prepared by the NMC and independently reviewed by a Qualified Nuclear Engineer (QNE) and two Senior Reactor Operators (SROs) by approximately 1030. Investigation of the suspect fuel configuration revealed that this configuration was not specifically allowed in the Spent Fuel Pool criticality analyses, so a Problem Investigation Form was completed at 1215. Repositioning of the suspect fuel assemblies was completed before this time.

The vendor responsible for the current Spent Fuel Pool Criticality Analysis was contacted to establish whether the suspect configuration was bounded by the existing analysis. The vendor responded that the suspect fuel configuration did not meet the initial assumptions made for the Spent Fuel Pool Criticality Analysis, and immediately began preparing an analysis of the safety impact of the suspect configuration.

A copy of the analysis indicated there was no safety significance of this fuel positioning other than requiring a minimum boron concentration in the Spent Fuel Pool of 300 PPM, which was exceeded at all times during this event.

This event is being reported pursuant to 10CFR50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications.

C. CAUSE OF EVENT:

The causes of the event were determined to be personnel error and procedural and management deficiencies.

The Nuclear Material Custodian and one Independent Reviewer did not identify the suspect fuel positioning during preparation of fuel movement plan. Although no known requirements for the placement of fuel at this transition

C. **CAUSE OF EVENT (continued):**

boundary between fuel meeting the burnup-initial enrichment criteria and fuel not meeting the criteria (stored in Region 2) of the Spent Fuel Pool had been transmitted to Braidwood Station by the analysis vendor, the NMC and the Independent Reviewer are expected to identify such a questionable configuration prior to NCTL issuance.

One Independent Reviewer of the prepared NCTLs identified the suspect fuel positioning as questionable. However, the reviewer did not address the question prior to approving the NCTLs.

The required fuel positioning at the interface between fuel that does meet the burnup-initial enrichment restriction and fuel that does not meet the criteria in Region 2 of the Spent Fuel Pool was not specified in any Braidwood Station or Commonwealth Edison procedures directing fuel movements.

The requirement for positioning fuel with less than or equal to 4.2 weight percent Uranium 235 that does not meet the burnup-initial enrichment criteria in a checkerboard configuration was transmitted by the "Licensing Report On High Density Spent Fuel Racks For Braidwood Units 1 and 2", Revision 0, dated August, 1988. This document addresses the assumptions made for the analysis, but does not identify any interface requirements.

The expectation to review the planned fuel movements against positioning requirements was not clearly defined. Inclusion of all requirements into fuel movement planning, and actual preparation of NCTLs for all types of fuel movement planning did not address these activities in sufficient detail.

The planning and independent review of the controlled NCTLs were performed using unverified and uncontrolled information.

D. **SAFETY ANALYSIS:**

There were no safety consequences for this event. Analysis by the vendor performing the Spent Fuel Pool Criticality Analysis indicates that the mispositioned fuel did not cause a criticality concern as long as sufficient

NRC FORM 366A (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

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LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Braidwood Unit 1	05000456	96	--007--	00	5 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

D. SAFETY ANALYSIS (continued):

boron existed in the Spent Fuel Pool. The required concentration for this event is 300 PPM. Spent Fuel Pool boron concentration remained in excess of 2300 PPM for the duration of this event. If a fuel mispositioning or fuel drop event had occurred while the fuel was mispositioned, sufficient boron concentration existed to maintain the Spent Fuel Pool in a safe condition.

E. CORRECTIVE ACTIONS:

Immediate corrective actions were to reposition the fuel in the inappropriate configuration.

The NMC and Independent Reviewers were counseled regarding this failure to meet expectations.

A new procedure, BwAP 2364-3T3, has been created to list the requirements for fuel positioning. Procedure changes have been generated to require execution of this new procedure prior to issuing NCTLs. The new procedure includes a checklist, requiring both the NCTL preparer and an independent verifier to review the proposed fuel movements for fuel positioning requirements.

The interface requirements for fuel storage that does meet the initial burnup-initial enrichment requirements were received from the analysis vendor. These requirements were reviewed against all other fuel stored in the Spent Fuel Pool. No other instances in which the requirements were not met were identified. These requirements were incorporated into Braidwood Station Procedures as BwAP 2364-3A1.

The Qualification Guide will be revised to provide more specific guidance regarding the necessity to review planned fuel movements against positioning requirements. This action will be tracked by NTS item #456-180-96-00701.

This event was discussed with all qualified Nuclear Engineers.

EXHIBIT B-4

Browns Ferry Unit 2:
Supplemental LER (0ctober 9, 1980)

TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
1750 Chestnut Street Tower II

October 9, 1980

Mr. James P. O'Reilly, Director
U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Dear Mr. O'Reilly:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 2 - DOCKET NO. 50-260 - FACILITY OPERATING LICENSE DPR-52 - REPORTABLE OCCURRENCE REPORT BFRO-50-260/8037 REVISION 1

The enclosed report is a supplement to my letter dated September 26, 1980, concerning fuel assemblies TZ 758 and TZ 399 which were misoriented 90 degrees. This report is submitted in accordance with Browns Ferry unit 2 Technical Specification 6.7.2.a(9).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

J. R. Calhoun
Director of Nuclear Power

Enclosure (3)
cc (Enclosure):
Director (3)
Office of Management Information and Program Control
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Director (40)
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Bill Lavallee
Nuclear Safety Analysis Center
Palo Alto, California 94303

Mr. R. F. Sullivan, NRC Inspector, Browns Ferry

8010150 513 S A003 1/1

NRC FORM 366 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

UPDATE REPORT - PREVIOUS REPORT September 26, 1980

LICENSEE EVENT REPORT

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 | A L B R F 2 (2) | 0 0 - 0 0 0 0 0 - 0 0 (3) | 4 1 1 1 1 (4) | (5)
LICENSEE CODE | LICENSE NUMBER | LICENSE TYPE | CAT

CON'T 01 REPORT SOURCE L (6) 0 5 0 0 0 2 6 0 (7) 0 9 1 4 8 0 (8) 1 0 0 9 8 0 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 During EOC-3 fuel shuffling operations it was noted that fuel assemblies TZ 758 and
03 TZ 399 were misoriented 90°. There was no previous occurrences. There was no
04 danger or hazard to the public. See Technical Specification 2.1 and 3.5.K.
05
06
07
08

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
09	R C (11)	A (12)	A (13)	F U E L X X (14)	Z (15)	Z (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 0	—	0 3 7	/	0 1	X	—	1

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	G (19)	Z (20)	Z (21)	0 0 0 0	Y (23)	N (24)	N (25)	G 0 8 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 During the previous refueling outage (BOC-3) 16 fuel assemblies were misoriented.
11 Subsequent rework left two 7 x 7 bundles in the misoriented position. A later
12 review of the BOC-3 core verification tapes confirmed there were no other misoriented
13 fuel assemblies. Units 1 and 3 core verification tapes will be verified.
14

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
15	H (28)	0 0 0 (29)	NA	A (31)	Operator observation

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
16	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
17	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
18	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
19	Z (42)	NA

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
20	Y (44)	TVA media information telephone	

NAME OF PREPARER 8010150 525 PHONE:

Tennessee Valley Authority
Browns Ferry Nuclear Plant

Form BF-17
BF 15.2
1/10/79

LER SUPPLEMENTAL INFORMATION

BFRO-50- 260 / 8037 Technical Specification Involved 2.1 & 3.5.K

Reported Under Technical Specification 6.7.2.a(9)

Date of Occurrence 9/14/80 Time of Occurrence 1900 Unit 2

Identification and Description of Occurrence:
Fuel assemblies TZ 758 in core location 15-26 and TZ 399 in core location 29-28 were found to be rotated 90° from their correct orientation.

Conditions Prior to Occurrence:
Unit 1 - 1055 MWe

Unit 2 - refuel shutdown

Unit 3 - Shutdown maintenance outage

Action specified in the Technical Specification Surveillance Requirements met due to inoperable equipment. Describe.
NA

Apparent Cause of Occurrence:
The 16 misoriented fuel assemblies were loaded out of proper orientation and core verification procedures detected the errors. Rework instructions failed to accomplish the required orientation of the two fuel assemblies.

Analysis of Occurrence:

See attachment

Corrective Action:

Verification and reorientation procedure for fuel loading verification have been made. Procedural changes include the requirements that rework will be documented with second party verification.

Failure Data:

NA

*Retention: Period - Lifetime; Responsibility - Administrative Supervisor

*Revision:

According to Supplemental Reload Licensing document NEDO-24169A the limiting fuel loading error is a rotated 8 x 8 (8D274) fuel assembly and assumes a rotation of 180°. The MCPR for the limiting event is the safety limit of 1.07. Any other misorientation would result in an MCPR greater than 1.07. The misoriented fuel assemblies were both high exposure - original 7 x 7 (7D250). Since both subject fuel assemblies were not of the limiting type, were sufficiently separated to prevent interaction, and since there were no significant transients during the cycle, the safety limit of 1.07 MCPR was not exceeded. Of the two fuel assemblies, the process computer indicates that TZ 758 at location 15-26 made the closest approach to its operational limit for 7 x 7 fuel of 1.33 MCPR on the following three occasions:

7/10/79 MCPR = 1.40

1/1/80 MCPR = 1.39

6/30/80 MCPR = 1.40

All operation was within the bounds of the reload licensing submittal. Both of the fuel assemblies are scheduled to be removed and will not be reloaded for BOC-4.

EXHIBIT B-5

Byron Station:
LER 454/96-008-00 (June 25, 1996)

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(4-95)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
BYRON NUCLEAR POWER STATION	05000454	1 OF 9

TITLE (4)

Fuel Assemblies Located in Incorrect Region of Spent Fuel Pool

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER
05	28	96	96	008	00	06	25	96		05000
										05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

OPERATING MODE (9)	5								
POWER LEVEL (10)	0		20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
			20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David D. Goff, System Engineer X2154	815-234-5441

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).		NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On 28 May, 1996, Byron Station nuclear engineers confirmed that fuel assemblies F37E, F44E, and G67F were residing in Region 2 of the Spent Fuel Pool (SFP) without meeting the requirements of Technical Specification (TS) 5.6.1.1.b.2, "Fuel Storage - Region 2." The assemblies did not meet the minimum burnup requirements, nor were they checkerboarded. The required minimum burnups were 32651 MWd/MTU, 32651 MWd/MTU, and 32771 MWd/MTU respectively. The actual burnups were 32648 MWd/MTU, 32638 MWd/MTU, and 32728 MWd/MTU respectively.

The cause of this event was cognitive personnel error. The computer spreadsheet used to verify minimum required burnup contained erroneous information for assemblies F37E, F44E, and G67F, and the data in the spreadsheet had not been independently verified. Personnel approving placement of G67F into SFP Region 2 did not have the current revision of Burnup criteria for determination of fuel assembly eligibility for placement into Region 2. Ultimately, the fuel assemblies' burnups were not verified to meet the requirements of TS 5.6.1.1 Amendment 68, "Fuel Storage - Criticality," prior to its implementation.

On 29 May, 1996, the three fuel assemblies were moved into Region 1, as allowed by TS 5.6.1.1.a.2, "Fuel Storage - Region 1." All fuel assemblies remaining in Region 2 were verified either to meet the minimum required burnup or to be stored in a checkerboard pattern.

This event resulted in no safety concerns. The event was bounded by both the older and the newer criticality analyses for Region 2 fuel storage. Adequate reactivity controls were in place to ensure that the k_{eff} limit of 0.95 required by TS 5.6.1.1, "Fuel Storage - Criticality" was not challenged during this event.

This event is reportable under 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's TS.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
BYRON NUCLEAR POWER STATION	05000454	96 -	008 -	00	2 OF 9

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 05-28-96 / 1700

Unit 1 Mode 5 - Cold Shutdown Rx Power Shutdown RCS [AB] Temperature/Pressure 84°F / 0 psig

Unit 1 Mode 4 - Hot Shutdown Rx Power Shutdown RCS [AB] Temperature/Pressure 335°F / 321 psig

B. DESCRIPTION OF EVENT:

Byron Administrative Procedure (BAP) 2000-3T1, "Spent Fuel Burnup Verification Checklist," is a checklist used to verify that fuel assemblies either have or have not accrued the minimum required burnup for uncheckerboarded SFP Region 2 storage. The minimum required burnup is calculated by linear interpolation between values given in BAP 2000-3A1, "Minimum Required Burnup as a Function of Enrichment for Region II High Density Spent Fuel Storage Racks." The values in BAP 2000-3A1 are intended to bound TS Figure 5.6-1, "Minimum Burnup Versus Initial Enrichment For Region 2 Storage."

On 10 February, 1993, Byron Station nuclear engineers (engineers 1 and 2) completed BAP 2000-3T1 for fuel assemblies including F37E and F44E. The checklist showed both assemblies with an initial enrichment of 3.8 wt% U-235 and a minimum required burnup for placement into Region 2 of 32540 MWd/MTU, given by BAP 2000-3A1 Rev 1. F37E and F44E had accrued actual burnups of 32648 MWd/MTU and 32638 MWd/MTU respectively. The minimum value of 32540 MWd/MTU was appropriate for an initial enrichment of 3.8 wt% U-235, and both assemblies met the Technical Specification requirement for uncheckerboarded Region 2 storage.

On 11 February, 1993, Nuclear Fuels Services (NFS) issued letter NFS:PSS:93-060 which, in part, stated that fuel assemblies F37E and F44E met the minimum burnup requirements of TS 5.6.1.1. This letter showed F37E and F44E having accumulated 32648.0 MWd/MTU and 32638.4 MWd/MTU respectively.

On 18 August, 1993, Byron Station fuel handlers moved fuel assemblies F37E and F44E into SFP locations K-C2 and K-D8, respectively, in Region 2. The assemblies were not stored in a checkerboard pattern since they met the minimum required burnup restrictions presently in place. The moves were performed in accordance with page 93-104 of an approved BAP 2000-3T3 Rev 1, "PWR Station Nuclear Component Transfer List." Engineers 1 and 3 verified that BAP 2000-3T1 was completed prior to transfer list approval.

Starting in the summer months of 1994, engineer 3 was assisting in the preparation of a license amendment request. This request would allow storage of fuel in Region 2 up to 5.0 wt% U-235 and was supported by a new criticality analysis.

On 11 August, 1994, Byron Station engineers (engineers 3 and 4) initiated Problem Identification Form (PIF) 454-201-94-69200. This PIF documented that Byron Station and NFS employed different methods in determining whether a fuel assembly meets the minimum burnup requirement for Region 2 storage. NFS used a polynomial fit through the points given in the criticality analysis after applying a 1.03 multiplicative penalty to account for fit error and uncertainty in the assembly burnup calculation. Byron Station used linear interpolation between points which bound TS Figure 5.6-1 Amendment 25. This PIF also identified that TS Figure 5.6-1 Amendment 25 did not, for all initial enrichments, bound the criticality analysis used as the basis for the curve.

B. DESCRIPTION OF EVENT (cont.)

Byron Station and NFS continued to use different criteria for minimum required burnup determination. The license amendment request being developed, when approved, would render the second problem moot. For the interim, engineer 3 prepared a revision request for BAP 2000-3A1 to change the points used for minimum burnup determination such that both TS Figure 5.6-1 Amendment 25 and the criticality analysis would be bounded.

On 16 September, 1994, Byron Station nuclear engineers (engineers 5 and 6) completed BAP 2000-3T1 for fuel assemblies including G67F. This checklist showed the G67F assembly with an initial enrichment of 3.809 wt% U-235 and meeting the minimum required burnup for placement into Region 2 of 32661 MWd/MTU. G67F had accrued an actual burnup of 32728 MWd/MTU. The minimum value of 32661 MWd/MTU was conservative for an initial enrichment of 3.809 wt% U-235. Engineer 6 stated that the enrichment value was conservatively rounded up to 3.81 wt% U-235 when the minimum required burnup was calculated. G67F met the Technical Specification requirement for uncheckerboarded Region 2 storage.

Also on 16 September, 1994, NFS issued letter NFS:PSS:94-225 which, in part, stated that fuel assembly G67F did not meet the minimum burnup requirements of TS 5.6.1.1. The discrepancy between the Byron Station and NFS conclusions resulted from the different methods in determining eligibility of a Region 2 storage candidate. Since G67F had accrued the minimum required burnup in accordance with BAP 2000-3A1 Rev 1, it was deemed to be suitable for uncheckerboarded Region 2 storage.

On 20 October, 1994, Byron Station Onsite Review (OSR) 94-078 approved a license amendment request for Byron Station Units 1 and 2 Technical Specifications. This amendment request later became TS Amendment 68. This request would, in part, revise Figure 5.6-1 Amendment 25 to be conservativ[illegible] 3% greater than the new criticality analysis. Discrete values would be provided in Figure 5.6-1 along with [illegible]structions that would allow linear interpolation between the values. In particular, the required burnup for an initial enrichment of 3.8 wt% U-235 would be increased from 32540 MWd/MTU to 32651 MWd/MTU.

The OSR 94-078 package did not document the review of incumbent fuel assemblies and their eligibility for Region 2 storage with the new minimum burnup curve. Engineer 3 and a representative from NFS participated in the OSR.

However, Byron Station nuclear engineers (engineers 3 and 7) had conducted a review of the incumbent fuel assemblies over the course of several months from approximately August to November, 1994. This review was performed by engineer 7 building a computer spreadsheet to calculate assembly eligibility, and then the ouput was spot checked by engineer 3 for verification. The spreadsheet required input data for initial enrichment, storage location, and actual accrued burnup, and then checked each fuel assembly against several minimum burnup criteria, including those that would become BAP 2000-3A1 Rev 2 and TS Amendment 68. The spreadsheet calculation produced a Boolean output for each assembly, i.e., "OK" or "not OK" for uncheckerboarded Region 2 storage.

Initial enrichment, storage location, and actual accrued burnup data loaded into the spreadsheet for F37E, F44E, and G67F were incorrect. This resulted in the spreadsheet producing erroneous "OK" outputs for those assemblies. Had correct data been loaded into the spreadsheet, the assemblies would have been properly identified as "not OK" when compared against the minimum required burnups of BAP 2000-3A1 and TS Amendment 68.

B. DESCRIPTION OF EVENT (cont.)

On 26 October, 1994, PIF 454-201-94-69200 was closed with the understanding that Byron Station and NFS would continue to use different methods for determining minimum required burnup for Region 2 storage. This would serve as a diverse means to identify assemblies suitable for Region 2 storage.

On 13 December, 1994, Byron Station OSR approved revision 2 of BAP 2000-3A1. This revision was processed as a corrective action to PIF 454-201-94-69200, which identified that TS Figure 5.6-1 Amendment 25 did not, for all initial enrichments, bound the criticality analysis used as the basis for the curve. The new revision bounded both the criticality analysis and TS Figure 5.6-1 Amendment 25. Under the new revision, the minimum required burnup for an initial enrichment of 3.8 wt% U-235 was increased from 32540 MWd/MTU to 32800 MWd/MTU. Byron Station took credit for the review performed in association with OSR 94-078 to verify compliance of the incumbent fuel assemblies. As stated before, the spreadsheet contained erroneous data for F37E, F44E, and G67F. Hence, all three assemblies passed the review. Under BAP 2000-3A1 Rev 2, fuel assemblies F37E, F44E, and G67F no longer met the minimum required burnup, though they all met the requirements of revision 1.

On 20 January, 1995, the Nuclear Regulatory Commission (NRC) issued Amendment 68 to Byron Station Units 1 and 2 TS, revising Figure 5.6-1 as requested under the licensing amendment request previously submitted.

On 23 January, 1995, Byron Station fuel handlers moved fuel assembly G67F into SFP location G-L12 in Region 2. The assembly was not stored in a checkerboard pattern since it had been verified to meet the requirements of BAP 2000-3A1 Rev 1. This was done in accordance with page 95-5 of an approved PWR Station Nuclear Component Transfer List. Engineers 5 and 8 verified that BAP 2000-3T1 Rev. 1 was completed prior to transfer list approval. However, BAP 2000-3T1 Rev. 1 had been completed in September, 1994, using BAP 2000-3A1 Rev 1. BAP 2000-3A1 Rev. 2 was now the current revision, and assembly burnups should have been compared to revision 2 requirements rather than the revision 1 requirements. The assembly did not meet the minimum burnup requirement of BAP 2000-3A1 Rev 2 or TS Amendment 68, though it did comply with TS Figure 5.6-1 Amendment 25.

On 25 January, 1995, Byron Station OSR 95-007 approved for use Amendment 68 and its implementation plan. The OSR 95-007 package acknowledged that TS Figure 5.6-1 was changing. The implementation plan stated that the Byron Station nuclear engineering group "will revise BAP 2000-3A1 to reflect the new burnup curve to identify assemblies that are acceptable to load in Region 2." At that time, it was thought that BAP 2000-3A1 Rev 2 was more conservative than TS Figure 5.6-1 Amendment 68. Therefore, the implementation plan required no deadline for revision of BAP 2000-3A1. The OSR package did not discuss the review that had been performed of the incumbent assemblies. Engineer 5 and the Station Reactor Engineer (SRE) participated in the OSR.

On 30 January, 1995, Byron Station OSR approved revision 3 of BAP 2000-3T2, "NCTL Verification Checklist." This revision provided more explicitly detailed guidance on how to perform the verification of minimum required burnups on BAP 2000-3T1.

On 8 February, 1995, Byron Station OSR approved revision 2 of BAP 2000-3T1. This revision added more documentation of information so that minimum required burnups could be more readily and accurately determined.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

B. DESCRIPTION OF EVENT (cont.)

On 1 March, 1995, all TS manual holders were instructed, in a letter from the Byron Station Regulatory Assurance Department Supervisor, to implement TS Amendments 67, 68, and 69. At this time, assemblies F37E, F44E, and G67F, were in Region 2 and were in violation of TS 5.6.1.1. Each had been previously approved for residence in Region 2 using a revision of BAP 2000-3A1 which reflected an earlier TS amendment.

On 17 August, 1995, Byron Station OSR approved revision 3 of BAP 2000-3A1. This revision was processed due to TS Amendment 68 changing the minimum required burnup curve. The procedure now exactly matched TS Figure 5.6-1, requiring 32651 MWd/MTU for an initial enrichment of 3.8 wt% U-235. Again, Byron Station took credit for the review performed in association with OSR 94-078 to verify compliance of the incumbent fuel assemblies. Two fuel assemblies were moved into SFP Region 2 since implementation of TS Amendment 68 on 1 March, 1995. They were moved from failed fuel canisters on 1 June and 29 June. Both assemblies met the minimum burnup requirement.

On 24 May, 1996, while performing BAP 2000-3T1 for fuel assemblies anticipated to be moved in association with upcoming spent fuel storage rack neutron attenuation testing, Byron Station nuclear engineers (engineers 7 and 9) found indications that fuel assemblies F37E and F44E did not meet the minimum burnup as required by TS 5.6.1.1.b.2.a, "Fuel Storage - Region 2." Nor were these two assemblies stored in a checkerboard pattern as allowed by TS 5.6.1.1.b.2.b, "Fuel Storage - Region 2." Byron Station contacted NFS for verification of actual burnup and minimum required burnup and to assist the investigation into whether these fuel assemblies were incorrectly residing in Region 2.

On 26 May, 1996, while performing BAP 2000-3T1 for fuel assemblies anticipated to be moved in association with upcoming spent fuel storage rack neutron attenuation testing, Byron Station nuclear engineers (engineers 7 and 9) found indications that fuel assembly G67F did not meet the minimum burnup as required by TS 5.6.1.1.b.2.a. Nor was this assembly stored in a checkerboard pattern as allowed by TS 5.6.1.1.b.2.b. Byron Station again contacted NFS for verification of actual burnup and minimum required burnup and to include this fuel assembly in the investigation.

On 28 May, Byron Station nuclear engineers (engineers 7, 9 and the acting SRE) and NFS held a conference call discussing the results of the NFS investigation into fuel assemblies F37E, F44E, and G67F. It was determined at 17:00 that all three assemblies were in violation of TS 5.6.1.1.b.2.

C. CAUSE OF EVENT:

The cause of F37E and F44E being incorrectly stored in Region 2 was cognitive personnel error. The data used by the computer spreadsheet for verifying minimum required burnup was not entered correctly nor was it independently verified to be accurate. The spreadsheet data failed to show that F37E and F44E were in SFP Region 2. Furthermore, the spreadsheet data failed to use the correct burnup values for F37E and F44E. This resulted in assemblies F37E and F44E producing erroneous "OK" spreadsheet outputs. This faulty technical review was part of the basis for the Byron Station OSR 95-008 approval and acceptance of TS Amendment 68. The amendment was then implemented with plant conditions not conforming to the new requirements.

NRC FORM 366A
(4-95)

U.S. NUCLEAR REGULATORY COMMISSION

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C. CAUSE OF EVENT (cont.)

The cause of G67F being incorrectly stored in Region 2 was also cognitive personnel error. Personnel approving the NCTL to place G67F in SFP Region 2 failed to use the current procedure revision of BAP 2000-3A1 to verify that G67F had accrued the minimum required burnup for uncheckerboarded Region 2 storage. The previous revision that was used did not reflect current plant conditions. This resulted in an ineligible fuel assembly being placed into Region 2.

D. SAFETY ANALYSIS:

The SFP condition throughout this event was bounded by the two criticality analyses used as the bases for TS Figure 5.6-1 prior to and after Amendment 68. All uncheckerboarded fuel assemblies, including F37E, F44E, and G67F, met the minimum burnup requirements of those analyses. However, the SFP condition failed to meet the current TS requirement, which was 3% greater than the current criticality analysis.

UFSAR section 9.1.3.2 addresses the safety evaluation for storing spent fuel in the SFP. The criticality portion is based on the "Byron and Braidwood Spent Fuel Rack Criticality Analysis Considering Boraflex Gaps and Shrinkage" document from Westinghouse dated June, 1994, as amended by 94CB*-G-0105 and 94CB*-G-0142. Section 5.0, Discussion of Postulated Accidents, addresses an abnormal condition where reactivity would increase beyond the analyzed condition: a fuel assembly is misloaded into Region 2 which does not satisfy the requirements.

While, in the scenario considered, only one assembly is misloaded, the analysis makes several conservative assumptions:

1. All fuel assemblies contain U-235 at the nominal enrichment or its equivalent at the minimum required burnup.

2. All fuel assemblies are uniformly enriched. No credit is taken for reduced-enrichment or natural uranium axial blankets.

3. No credit is taken for U-234, U-236, or any fission product poisons. No credit is taken for any burnable absorber material which may remain in the fuel.

4. All storage locations are loaded with fuel assemblies not containing any absorption material.

5. The storage locations are infinite in lateral extent.

6. The array is moderated by pure water of 1.0 g/cc.

7. A conservative Boraflex degradation model is assumed.

8. The scenario where a fresh assembly with an enrichment of 4.2 wt% is inserted into a 5x5 array of the nominal assemblies is considered.

D. Safety Analysis (cont.)

The maximum k_{eff} at a 95% probability with 95% confidence and including the statistical summation of independent uncertainties is 0.9449 for Region 2 under the nominal conditions. The increase in reactivity due to the misloaded assembly is no more than 0.0438 delta k. However, only a single failure must be accounted for, so soluble boron may be credited. The reactivity from 300 ppm boron is approximately -0.06 delta k, more than offsetting the increase from the misloading. Thus, the k_{eff} limit of 0.95 required by TS 5.6.1.1 is not challenged during this abnormal condition.

The situation described in this report, with three fuel assemblies misloaded rather than just one, is more conservative than the accident analysis due to the following considerations:

1. Nearly all fuel assemblies residing in Region 2 exceed the minimum burnup requirement, making them less reactive than the reference assemblies.

2. Many fuel assemblies have reduced-enrichment or natural uranium axial blankets of six inches at both ends, reducing their reactivities.

3. All fuel assemblies contain U-234 and U-236, and spent assemblies contain fission product poisons as well. These materials further reduce reactivity.

4. Not every storage location contains fuel. Locally, there are several empty locations. Some of the fuel assemblies contain absorber material such as rod cluster control assemblies (RCCAs).

5. The SFP is finite, exhibiting nonzero neutron leakage at the boundaries.

6. The water in the SFP is normally approximately 80 degF, having a density less than 1.0 g/cc. Soluble boron concentration in the SFP remained greater than 1280 ppm since January, 1995, providing at least -0.22 delta k reactivity.

7. Previous neutron attenuation testing results imply that the Boraflex in Region 2 has not deteriorated to the extent assumed in the analysis.

8. The improperly located fuel assemblies are significantly less reactive than the fresh 4.2 wt% enriched assembly assumed in the accident analysis. Fuel assemblies F37E, F44E, and G67F fell short of the required burnup by 3 MWd/MTU, 13 MWd/MTU, and 43 MWd/MTU respectively. These values are within approximately 0.1% of the required burnup values.

The combination of the above factors ensured that the k_{eff} limit of 0.95 required by TS 5.6.1.1 was not challenged during this event.

E. CORRECTIVE ACTIONS:

On 28 May, 1996, at 17:15, Byron Station nuclear engineers initiated PIF 454-180-96-0008, identifying three fuel assemblies inappropriately residing in Region 2 of the SFP. Byron Station Regulatory Assurance, Operations, and System Engineering management were notified. The NRC Resident Inspector was also notified.

Concurrently, NFS initiated PIF 901-201-96-07800 identifying possible inadequacies and inconsistencies in their methods of determining eligibility of Region 2 candidate fuel assemblies. The investigation results show that these inadequacies and inconsistencies did not contribute to the root causes of this event.

On 29 May, 1996, at 05:15, Byron Station fuel handlers moved fuel assemblies F37E, F44E, and G67F into SFP storage locations in Region 1. This was done in accordance with page 96-103 of an approved PWR Station Nuclear Component Transfer List.

NFS subsequently performed a review of all fuel assemblies residing in Region 2 using TS Amendment 68 criteria. This review was transmitted as NFS:PSS:96-142 and PSSCN:96-023. It consisted of a list of every fuel assembly in the Byron Station SFP as of 31 March, 1996, and identified which assemblies had achieved the minimum required burnup for Region 2 storage. Byron Station engineers 7 and 9 then verified that those assemblies not meeting minimum burnup were either stored in Region 1 or in a checkerboard pattern. There were no assemblies stored inappropriately in Region 2. All fuel moves into Region 2 performed since 31 March, 1996, have had eligibility requirements verified in accordance with BAP 2000-3A1 Rev 3.

BAP 2000-3T2 Rev 3 is currently in place and provides explicit guidance on the preparation and independent review of BAP 2000-3T1 Rev. 2. This revision was not in place at the times F37E, F44E, and G67F were approved for uncheckerboarded Region 2 storage. The guidance provided presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

BAP 2000-3T1 Rev. 2 is currently in place and provides improved documentation of minimum required burnup for fuel assemblies being moved to or within Region 2. This revision was not in place at the times F37E, F44E, and G67F were approved for uncheckerboarded Region 2 storage. The improved documentation shows initial enrichment, minimum required burnup, and actual accrued burnup for each assembly and presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

BAP 2000-3A1 Rev. 3 is currently in place and is identical to the requirements of TS Figure 5.6-1 Amendment 68 as well as the current NFS method of determining Region 2 storage eligibility. All future fuel assemblies approved for Region 2 storage will have minimum required burnups determined in accordance with this procedure or its equivalent. Any future TS Amendment changing TS Figure 5.6-1 will have a concurrent revision to BAP 2000-3A1 associated with it reflecting the new requirements. This presents an additional barrier to mislocating a fuel assembly that could have prevented this event.

Performance expectations have been discussed with persons involved in the errors that contributed to this event.

This LER will be discussed with all members of the Byron Station nuclear engineering group, emphasizing personnel performance expectations. A copy will be placed in the nuclear engineering group required reading book. NTS item 454-201-96-0008-01 tracks completion of this action.

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F. RECURRING EVENTS SEARCH AND ANALYSIS:

LER 454:94-006, "Fuel Assembly Located in Wrong Region of Spent Fuel Pool due to Personnel Error," documents a similar event. On 15 July, 1994, SED found a fuel assembly in Region 2 that neither met the minimum burnup requirements of TS Figure 5.6-1 nor was checkerboarded. The cause of this event was determined to be cognitive personnel errors. The Nuclear Materials Custodian and an independent reviewer failed to use the approved method to verify assemblies met the minimum burnup requirements for storage in Region 2.

Although the 454:94-006 event resulted in a fuel assembly incorrectly residing in SFP Region 2, the circumstances leading to this event were different from those leading to the 454-180-96-0008 event.

G. COMPONENT FAILURE DATA:

No components failed in association with this event.

EXHIBIT B-6

Byron Station:
LER 454/94-006-00 (August 15, 1994)

SIGNATURE PAGE FOR LICENSE EVENT REPORT

LER Number
454 : 94-006

Title of Event: Fuel Assembly in Wrong Location Spent Fuel Pool due to Personnel Error

Occurred: 09-15-94/ 0930
Date Time

OSR DISCIPLINES REQUIRED: ABCG

SES / 9/13/94 DATE

Acceptance by Station Review:

9/12/94		9-15-94	
OE	Date	SES - ABG	Date
9/15/94		9/15/94	
RAS	Date	OTHER - CG	Date

Approved by 9/15/94
Station Manager Date

09319.WPF 082994-2

LICENSEE EVENT REPORT (LER)

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TITLE

FUEL ASSEMBLY LOCATED IN WRONG REGION OF SPENT FUEL POOL DUE TO PERSONNEL ERROR

EVENT DATE			LER NUMBER			REPORT DATE			OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQ. NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 7	1 5	9 4	9 4	- 0 0 9 -	0 0	0 8	1 5	9 4	Byron Unit 2	0 5 0 0 0 4 5 5
										0 5 0 0 0

OPERATING MODE: 1

POWER LEVEL: 1 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (CHECK ONE OR MORE OF THE FOLLOWING)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER
G. STAUFFER, STATION REACTOR ENGINEER, X2249	8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines)

On July 15, 1994, System Engineering Department (SED) found fuel assembly U38J located in Region II of the Spent Fuel Pool (SFP). The fuel assembly did not meet the burnup requirements specified in Technical Specifications (TS) Section 5, "Design Features," Figure 5.6-1, "Minimum Burnup Versus Initial Enrichment for Region II Storage." The Nuclear Component Transfer List (NCTL) incorrectly specified the placement of U38J into Region II at location HM5. The NCTL also did not place the assembly into Region II in a checkerboard pattern. Administrative controls require any assembly that does not meet minimum burnup to be placed into Region II in a checkerboard pattern. The assembly was placed into the incorrect region of the SFP on September 26, 1993 during a refueling outage on Unit 2.

The error was discovered while preparing for the next refueling outage. The assembly was moved to Region I on July 16, 1994.

This event involved no safety concerns. The safety significance of the misplaced assembly is within the safety analysis presented in the UFSAR. This event is reportable in accordance with 10CFR 50.73(a)(2)(i)(B). Any operation or condition prohibited by the plant's Technical Specifications

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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 07/15/94 / 0930

Unit 1 MODE 1 - Power Operations Rx Power 80% in coastdown RCS [AB] Temperature/Pressure NOT/NOP

Unit 2 MODE 1 - Power Operations Rx Power 99% RCS [AB] Temperature/Pressure NOT/NOP

B. DESCRIPTION OF EVENT:

Between mid-August, 1993 and September 10, 1993, a non-licensed engineer (Engineer 1) completed the Nuclear Component Transfer Lists (NCTLs) for offloading the Unit 2 reactor core (Page numbers 93-121 to 93-146). This individual was the station's Nuclear Materials Custodian or NMC. During the writing of the NCTLs, he made two errors. On page 93-139, the NCTL shows fuel assembly U29J going to storage location HM10. This location is in a Region II rack. The burnup of the assembly, at the time the NMC wrote the list, did not meet the minimum burnup requirement for placement into Region II. The NMC made a similar mistake for fuel assembly U38J on page 93-143. The actual burnup of U38J was 29770 MWD/MTU versus a required burnup of 32,540 MWD/MTU. The NCTL shows assembly U38J going to storage location HM5. This location is also in a Region II rack. Both errors were cognitive personnel errors.

After the NMC wrote the NCTL for the offload, for Refueling Outage B2R04, he completed Byron Administrative Procedure (BAP) BAP 2000-3T2, "Nuclear Component Transfer List (NCTL) Verification Checklist." Step 1 of the checklist requires the preparer of the NCTL to verify that,

> "Fuel assemblies entering Region II of the spent fuel racks meet minimum burnup requirements as described in BAP 2000-3A1 or are placed into a checkerboard configuration. Records of assemblies which meet minimum burnup requirements are kept in file 1.02.1080, which is in the NMC satellite file cabinet."

Records of assemblies that meet minimum burnup are documented on BAP 2000-3-T1, "Spent Fuel Burnup Verification Checklist," and are kept in file location 1.02.1080. BAP 2000-3A1's title is, "Minimum Required Burnup as a Function of Enrichment for Region II High Density Spent Fuel Storage Racks." This attachment gives a listing of initial enrichment versus the minimum burnup required for storage in a Region II rack.

BAP 2000-3, "Safeguarding and Controlling Movements of Nuclear Fuel Within a Station," requires the NMC to complete BAP 2000-3-T1 for each assembly to be placed into Region II of the Spent Fuel Pit (SFP). The NMC started but did not complete these forms for assemblies placed into Region II during Outage B2R04. The BAP 2000-3-T1 form was completed as part of this investigation.

The NMC used the TOTE data for all the assemblies discharged from the core. TOTE is a computer program that calculates assembly burnup. TOTE data gives the total accumulated burnup for each fuel assembly. The data is stored on the IBM mainframe and is accessible via a personal computer. The NMC used the IBM and mentally went through the burnup verification. He did not complete the information on BAP 2000-3 T1. Nuclear Fuel Services (NFS) is responsible for running the code. They run the code every month and after a unit shutdown.

B. DESCRIPTION OF EVENT: (Cont.)

Using the TOTE burnup data and the initial enrichment of each assembly, the NMC did the burnup check using BAP 2000-3A1. The NMC could not recall why he did not complete the forms as required by procedure or why he made the error when he did the burnup checks. A review of the BAP 2000-3-T1 forms for the previous outage (February 1993, B1R05) showed the NMC had completed the forms.

Discussions with the NMC identified several weaknesses in the NCTL writing process. The process is very complicated and relies heavily on the skills of the individual writing the NCTLs. The Verification Checklist gives criteria that the NCTLs must meet. However, the checklist does not describe the process on "how to" write the NCTLs. The NMC divided the process into three major sequences: the offload, the insert shuffle, and the onload. The process as described by the NMC is given below.

First, the NMC does a comparison between the candidate loading pattern supplied by NFS, and the existing core loading pattern. The candidate loading pattern shows the next cycle's core loading pattern. The NMC obtained the existing loading pattern from the tagboard for the Unit 2 reactor core. The tagboards are located in the area where the NMC sits. The tagboards are used to show the location of every fuel assembly and component in the SFP, the New Fuel Storage Racks, Failed Fuel Storage Racks, and the two reactor cores. The tagboards mimic the physical layouts of each of these areas of the plant. And, the NMC keeps them up-to-date based on completed NCTLs.

Once he completed this comparison, he placed each assembly into categories. He based the categories on the insert a fuel assembly contained in the current cycle and the insert the fuel assembly would have in the next cycle. In other words, categories of assemblies are based on what they "have" and what they are "getting." For this event, there were nine different categories. For example, assemblies that have burnable poisons (BPs) that are getting thimble plugs (TPs) (BPs to TPs), assemblies that have control rods (RCCAs) and are getting thimble plugs (RCCAs to TPs), and assemblies that have thimble plugs and are getting control rods (TPs to RCCAs).

Next, the NMC arranged the categories side-by-side in the SFP such that the insert swaps can occur with the least amount of tool changes. There are five major steps to the insert shuffle.

The NMC did this arrangement in the SFP by iteration until he obtained the most efficient layout. After the arrangement in the SFP is done, the NMC can begin writing the offload. The NMC wrote the offload such that the fuel assemblies were placed into the first open location in each of the nine categories. As he wrote the offload sequence, the NMC also ensured that each step met seven requirements and three optional items.

The NMC went through a similar process to write the insert swaps and the core onload sequences. In all, there were eleven required checks and four desirable items for the entire refueling. During discussions, the NMC identified an additional four criteria he met while writing the NCTLs, that were not part of BAP 2000-3T2. This brought the total number of checks the NMC met to nineteen.

After the NMC wrote the three major sequences, they were loaded into a computer program called Shuffle Works. This program wrote the sequence on NCTL forms that the Fuel Handlers used in the field. A member of the SED nuclear group entered the offload and insert shuffle into the program step-by-step. This was done because Shuffle Works could not perform all of the required checks. However, the program did write the onload sequence since it contained 1) the pool configuration after the core was offloaded and all the insert shuffles were done, 2) the final core configuration, and 3) the loading sequence. Because the program had this information, it, by default, wrote the sequence meeting all the appropriate requirements.

B. DESCRIPTION OF EVENT: (Cont.)

After the NMC wrote the NCTLs, he gave them to an independent reviewer on September 10, 1993. The independent reviewer was a non-licensed engineer (Engineer 2). Engineer 2 did not use the records of assemblies that meet minimum burnup requirements to verify certain assemblies could be placed into Region II. He was unaware of the requirement because he failed to review BAP 2000-3 prior to performing the verifications. This was a cognitive personnel error. Instead, this individual used information from the Nuclear Fuel Services Department (NFS). NFS sent a letter that listed assemblies by region and indicated which assemblies met the minimum burnup requirement for storage in Region II. Attached to the letter, was a printout showing the individual burnups of every assembly.

During his review, Engineer 2 found the error for fuel assembly U29J on page 93-139, but failed to find the error for assembly U38J on page 93-143. He notified the NMC of the error for U29J and the NMC wrote a variation to the NCTL. Engineer 2 did not discover the second error and stated that the cause of the error was most likely due to his performing several checks simultaneously. At the time he reviewed the NCTLs, he was performing multiple checks as he went through the NCTLs. This probably caused him to miss the burnup check for assembly U38J. Engineer 2 and the NMC both signed the verification checklist on September 13, 1993.

Discussions with Engineer 2 indicated that there have been errors in past NCTLs but they had been caught by the independent reviewer. No Problem Identification Forms (PIFs) were written for these events. Although PIFs were not required for these events, opportunities to identify and correct these errors before a higher level event occurred, were missed.

Fuel Handlers placed assembly U38J into a Region II rack on September 26, 1993 in accordance with the NCTL.

On July 15, 1994, a non-licensed engineer (Engineer 3) discovered that fuel assembly U38J was in a Region II spent fuel rack. The fuel assembly had been in the Region II rack since September 26, 1993. The Fuel Handlers had placed the assembly in the Region II rack during the last refueling on Unit 2. The assembly did not meet the minimum burnup requirements of Technical Specification Figure 5.6-1, "Minimum Burnup versus Initial Enrichment for Region II Storage."

Engineer 3 discovered the error during preparations for moving fuel assemblies from Region I to Region II for the upcoming refueling outage on Unit 1 (B1R06). The SED Nuclear group reviewed every fuel assembly located in Region II to ensure the assemblies either met minimum burnup or were checkerboarded. After the discovery, Fue Handlers moved fuel assembly U38J to Region I following an approved Nuclear Component Transfer List (NCTL). The Fuel Handlers moved the assembly into Region I on July 16, 1994.

This event did not involve any inoperable systems and was not effected by plant operations on Unit 1 or 2. No operator actions either increased or decreased the severity of the event.

This event is reportable under 10 CFR 50.73(a)(2)(i)(B), any operation or condition prohibited by the plant's Technical Specifications

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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

C. CAUSE OF EVENT:

The primary causes of this event were cognitive personnel errors. Both the NMC and the independent reviewer failed to use the approved method to verify assemblies meet the minimum burnup requirements for storage in Region II racks. It should be noted that use of the approved method would not guarantee this mistake would not recur because of a procedural weakness. The procedure will be enhanced. There were also several contributing causal factors for this event that led to the cognitive personnel errors.

The current methodology for writing NCTLs is not well defined and relies heavily on the skills of the preparer. The preparer goes through many manual iterations on the NCTL until the most efficient sequence is found. This method is not conducive to minimizing human error.

The methods to be used for verification are also not well defined. Many verification steps required by BAP 2000-3T2 can be done in several different ways, as occurred during this event. And, some methods may not be as effective as others in catching errors or for performing verifications.

By not writing PIFs for failures found during independent verifications, the ability to find and correct problems <u>before</u> they result in higher level events such as an LER was minimized.

Although the Shuffle Works program is an effective program for its intended purpose, enhancement of the Shuffle Works program could help prevent errors of this type in the future.

A corrective action from a previous event was ineffective. Refer to the Recurring Events Search and Analysis section for an explanation.

D. SAFETY ANALYSIS:

UFSAR Section 9.1.2.3, "Safety Evaluation," says that "The largest reactivity increase occurs from accidentally placing a new fuel assembly into a Region II storage cell with all other cells fully loaded. Under this condition, the presence of 300 ppm soluble boron assures that the infinite multiplication factor would not exceed the design basis reactivity for Region II. With the recommended concentration of soluble poison present (2000 ppm boron), the maximum reactivity, K_{eff}, is less than 0.95 even if Region II were to be fully loaded with fresh fuel of 4.2% enrichment."

Byron Station normally maintains the boron concentration in the SFP at two thousand ppm and administratively controls the concentration to greater than eight hundred ppm. At the time it was placed into the SFP, fuel assembly U38J had a burnup of [illegible]9,770 MegaWatt-Days per Metric Ton-Uranium (MWD/MTU) and an initial enrichment of 3.802%. Therefore, the UFSAR analysis bounds the misplaced assembly and no safety significance existed while the assembly was in the Region II rack.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER			PAGE
		YEAR	SEQ. NUMBER	REVISION	
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 4	9 4	– 0 0 8 –	0 0	0 6 OF 0 7

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

E. CORRECTIVE ACTIONS:

Corrective Actions - Long Term

1. The NMC and the individual that performed the independent verification were counseled.

2. The SED nuclear group will write a procedure that explains the methodology to be used to write NCTLs for a refueling operation. In addition, this instruction will give directions on when in the process verifications will be done and the preferred method for performing the verifications. NTS item 454-180-94-00600-01 tracks completion of this item.

3. The SED Nuclear group will determine the preferred method for performing each verification on the Nuclear Component Transfer List (NCTL) Verification Checklist, BAP 2000-3-T1. SED will revise the checklist to:

   a) explicitly define the preferred method of each verification.
   b) indicate whether alternate methods are allowed and explicitly define these alternate methods. These methods will be equivalent to the preferred method.
   c) organize the checklist to distinguish important checks from less important checks.
   d) provide cautions describing the pitfalls for each method.

   NTS Item 454-180-94-00600-02 tracks the completion of these items.

4. The SED nuclear group will pursue revisions to the Shuffle Works program that will allow it to perform more of the verifications the nuclear group presently does manually. NTS item 454-180-94-00600-03 tracks completion of this item.

5. Regulatory Assurance will issue PIF threshold guidelines that will require writing PIFs for errors caught during independent reviews. NTS item 454-180-94-00600-04 tracks completion of this item.

6. The SED nuclear group will revise the BAP 2000-3-T1 form to include a column for recording both the assembly's burnup in addition to the minimum required burnup for storage in Region II. NTS item 454-180-94-00600-05 tracks completion of this item.

7. The SED nuclear group will revise BAP 2000-3 to require a walkthrough of the entire refueling on "paper" tagboards. NTS item 454-180-94-00600-06 tracks completion of this item.

Interim corrective actions for the upcoming refueling outage on Unit 1:

a. The Station Reactor Engineer will discuss this event with all members of the Nuclear Group and place this LER in the Nuclear Group Required Reading. NTS item # 454-180-94-00600-07 tracks this item.

b. BAP 2000-3-T1 will be used prior to moving any fuel into Region II. This is presently a requirement of BAP 2000-3, so no NTS item is needed to track this action.

c. A "paper" tagboard will be used for a step by step walkthrough of the entire refueling procedure. NTS item 454-180-94-006-08 tracks this item.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER YEAR	SEQ. NUMBER	REVISION	PAGE
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 4	9 4	– 0 0 8	– 0 0	0 7 OF 0 7

TEXT - Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

F. RECURRING EVENTS SEARCH AND ANALYSIS:

A search on ETS found one previous event of a misplaced fuel assembly due to an error in an NCTL. DVR 6-1-91-071, "Fuel Transfer List Error," documents this event. A review of the corrective actions for this event indicated that one of the corrective actions was not implemented. Corrective action to prevent recurrence, item 2B, states,

"BAP 2000-3 will be revised to require the use of a procedural checklist when developing the NCTL. This list will include:"

"B. The requirement to use a tag board."

Currently, BAP 2000-3 does not contain this requirement. Discussions with the Station Reactor Engineer (SRE), at the time of the event, indicated that this corrective action required a step by step walkthrough of the entire refueling evolution on the tagboards. However, Engineer 2 indicated that the intent of the corrective action changed. The intent changed to the use of a "paper" tagboard as opposed to the use of the physical tagboards. This would eliminate possible errors from moving chips on the physical tagboards. It cannot be determined why this requirement was not incorporated into BAP 2000-3. A review of the NTS item written to track completion of this corrective action indicated that the NTS was not specific on exactly what changes to BAP 2000-3 were needed. The NTS item simply stated to "develop a procedure checklist that specifies how to prepare an NCTL." At the time the checklist was developed, it failed to incorporate this requirement into BAP 2000-3. Therefore, this corrective action was ineffective.

G. COMPONENT FAILURE DATA:

There was no failed component during this event.

EXHIBIT B-7

Catawba Unit 1:
LER 413/90-0160-00 (April 19, 1990)

Duke Power Company
Catawba Nuclear Station
P.O. Box 256
Clover, S.C. 29710

(803) 831-3000

DUKE POWER

RECEIVED

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April 18, 1990

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station
Docket No. 50-413
LER 413/90-16

Gentlemen:

Attached is Licensee Event Report 413/90-16 concerning TECHNICAL SPECIFICATION VIOLATION AS A RESULT OF A MISSED REFUELING WATER STORAGE TANK SAMPLE DUE TO INAPPROPRIATE ACTION.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Tony B. Owen

Tony B. Owen
Station Manager

keb\LER-NRC.TBO

xc: Mr. S. D. Ebneter
Regional Administrator, Region II
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Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Mr. W. T. Orders
NRC Resident Inspector
Catawba Nuclear Station

IE22
1/1

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Catawba Nuclear Station, Unit 1

DOCKET NUMBER (2): 0 5 0 0 0 4 1 3

PAGE (3): 1 OF 0 9

TITLE (4): Technical Specification Violation As A Result Of A Missed Refueling Water Storage Tank Sample Due To Inappropriate Action

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 3	0 5	9 0	9 0	0 1 6	0 0	0 4	1 9	9 0	N/A	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
R.M. Glover, Compliance Manager	8 0 3	8 3 1 - 3 2 3 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

During the period of February 5 through 26, 1990, samples for the Boric Acid Tank (BAT) and the Refueling Water Storage Tank (FWST) were collected by Chemistry (CHM) to comply with Technical Specification (T/S) requirements. On February 5, CHM had been informed by Operation (OPS) personnel that the BAT was the declared borated water source. From March 11 through March 13, the FWST was not placed into recirculation and was not sampled due to the use of the Refueling Water (FW) pump for draining of the reactor cavity. On March 14, 1990, Unit 1 was in Mode 5, Cold Shutdown. CHM contacted the Control Room Operator (CRO) to verify that the BAT was still considered the declared borated water source. CHM was informed that the BAT had been inoperable since March 1, 1990 due to 1NV236B, being tagged out for repair. Following CHM review of data, during the week of March 5 through 12, 1990, CHM missed a T/S sample of the FWST. This event was attributed to inappropriate action, due to the individuals involved not ensuring an operable borated water source. A contributing cause is assigned to deficient communications resulting from poor group interface between CHM and OPS. Corrective actions taken included CHM procedure revisions which will supply actions to take when T/S samples cannot be obtained as well as including a T/S Operability Sheet for T/S items. Also, the above mentioned CHM corrective acitons will be communicated to OPS Shift personnel.

BACKGROUND

REFUELING WATER SYSTEM

The Refueling Water [EIIS:CB] (FW) System provides a large source of borated water and the necessary equipment to:

1. Supply the Emergency Core Cooling System (ECCS) and the Containment Spray [EIIS:BE] (NS) System during the injection phase following a Loss of Coolant Accident (LOCA);

2. Transfer the borated water between the Refueling Water Storage Tank (FWST) and Refueling Cavity;

3. Provide cleanup of the refueling water by routing the water through the Spent Fuel Pool Cooling [EIIS:DA] (KF) System; and,

4. Provide for various other borated water requirements and miscellaneous flowpaths.

The FWST normal capacity of 395,000 gallons is sufficient to provide a useable volume exceeding 350,000 gallons. This capacity assures:

a. The volume of borated refueling water needed to increase the boron concentration of initially spilled water to a level that assures no return to criticality with the Reactor at Cold Shutdown and all control rods [EIIS:ROD], except the most reactive Rod Cluster Control Assembly (RCCA), inserted in the core.

b. The volume of water sufficient to refill the Reactor vessel [EIIS:VSL] above the nozzles [EIIS:NZL] after a LOCA.

c. A sufficient volume of water when combined with ice melt and Reactor Coolant [EIIS:AB] (NC) System spill in the containment recirculation sump following a LOCA to permit the initiation of the recirculation phase.

d. A sufficient volume of water to limit the radiation dose rate at the surface of the Refueling Cavity to approximately 2.5 mrem/hr during the period when a fuel assembly is transferred over the Reactor vessel flange.

e. A sufficient volume of water to allow the station operator adequate time to complete the valve [EIIS:V] alignment required to complete the switchover from the injection mode to the containment sump recirculation mode following a LOCA.

When draining the FWST, the water is routed to the Refueling Cavity and to one of the Boron Recycle [EIIS:CA] (NB) System Recycle Holdup Tanks (RHTs). Approximately 290,000 gallons of water is drained to the Refueling Cavity while the remainder is drained through the KF purification loop into either one of the RHTs.

The refueling water from the Refueling Cavity is routed back to the FWST by using the normal refueling drain procedure. The water in the RHT is rerouted through the recycle evaporator feed pumps [EIIS:P] into the FWST. The water is brought back into specification by adding demineralized water or boric acid from the boric acid blender.

CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control [EIIS:CB] (NV) System is designed to provide the following services to the NC System:

1. Maintenance of programmed water level in the pressurizer.

2. Maintenance of seal-water injection flow to the NC pumps.

3. Control of water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup.

4. Filling, draining, and pressure testing.

The water chemistry, chemical shim and makeup requirements of the NC System are such that the following functions must be provided:

1. Means of addition and removal of pH control chemicals for Startup and normal operation.

2. Control of oxygen concentration following venting and that due to radiolysis in the core region during normal operation.

3. Means of purification to remove corrosion and fission products.

4. Means of addition and removal of soluble chemical neutron absorber and makeup water at concentrations and rates compatible with all phases of plant operation including emergency conditions.

The function of soluble neutron absorber concentration control and makeup is provided by the Reactor Makeup Control System employing 4 wt. percent boric acid solution from the Boric Acid Tank (BAT) and Reactor makeup water from the

Reactor Makeup Water Storage Tank (RMWST). In addition, for emergency boration and makeup the capability exists to provide refueling water or 4 wt. percent boric acid from the BAT to the suction of the charging pumps.

Two boric acid tanks are provided. The combined capacity of the tanks contains sufficient boric acid to provide for refueling plus enough boric acid for one Cold Shutdown immediately following refueling with the most reactive control rod withdrawn. There is sufficient capacity with one tank one-third full, to provide Cold Shutdown for the Unit with the most reactive rod withdrawn.

Technical Specification 3.1.2.5 states that as a minimum, one of the following borated water sources shall be OPERABLE (in MODES 5 & 6):

a. A Boric Acid Storage System with:

1. A minimum borated water volume of 5100 gallons,
2. A minimum boron concentration of 7000 ppm, and
3. A minimum solution temperature of 65 degrees F.

b. The Refueling Water Storage Tank with:

1. A minimum borated water volume of 26,000 gallons,
2. A minimum boron concentration of 2000 ppm, and
3. A minimum solution temperature of 70 degrees F.

T/S Surveillance Requirement 4.1.2.5 requires that the above borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid storage tank solution temperature when it is the source of borated water.

Chemistry procedures require sampling of the FWST once per week and sampling of the BAT twice per week.

EVENT DESCRIPTION

On February 5, 1990, Unit 1 was in Mode 6, Refueling. At 0630 hours, Chemistry (CHM) Technician A recorded in the Primary CHM logbook turnover notes that the Refueling Water Storage Tank (FWST) was in the process of makeup, and sampling was required. At 1400 hours, CHM Technician B telephoned the CRO to request that the FWST be placed in recirculation. The CRO informed the technician that makeup had stopped and that Operations (OPS) was concentrating on increasing the levels in the Boric Acid Tank (BAT). Due to the T/S requirement for once per seven day samples to be taken on either the BAT or the FWST, if the FWST was the "declared borated water source" then the sample would need to be taken no later than February 6. The OPS Shift Supervisor informed CHM Technician B that the BAT was the borated water source.

From February 6 through 11, 1990, at 1022 hours, Unit 1 was in Mode 6. All required FWST and BAT samples were collected and analyzed by CHM personnel.

From February 11 through 26, 1990, Unit 1 was in No Mode, Core Defueled. CHM personnel collected and analyzed all required BAT and FWST samples.

On February 24, 1990, Unit 1 was in No Mode. CHM Technician C was informed by the CRO that the 1B Residual Heat Removal [EIIS:BP] ND Pump was on and that the Reactor cavity water was being pumped back to the FWST. At approximately 1439 hours, Diesel Generator (D/G) 1B was removed from service, as a result of work list items related to the Outage.

Unit 1 entered Mode 6 on February 28, 1990. On March 1, 1990, at 0220 hours, Unit 1 remained in Mode 6. OPS issued R&R 19-2838 on 1NV236B, Boric Acid to NV Pumps Suction, for MOVATS testing and also issued R&R 10-807 on A and B Boric Acid Transfer Pumps for the 1NV236B work. This action in combination with D/G 1B being out of service necessitated the determination, by OPS that the BAT was inoperable, due to the unavailable BAT water source alignment. This change in BAT status was unknown by CHM. BAT sampling continued at the prescribed interval.

On March 4, 1990, at 0725 hours, Unit 1 was in Mode 6. CHM Technician C contacted the Unit 1 CRO to request that the FWST be placed in recirculation for the weekly sample. CHM Technician C was told that the FW pump was currently pumping down the Reactor cavity, and OPS was not able to state when the pump would be available. The CRO would check with the Shift Supervisor about the situation. CHM Technician C called the CRO again at 0832 hours, and there had been no determination made. At 1930 hours, CHM Technician D discussed the FWST status with the Unit Supervisor and was advised that the draining of the cavity had to be completed to permit FWST sampling.

On March 5, 1990, Unit 1 was in Mode 6. At 0050 hours, the weekly FWST T/S sample for boron analysis was due, but was not collected as a result of the FW pump being in service for Reactor cavity draining. The FWST was last sampled at 0050 hours on February 26.

On March 9, 1990, Unit 1 was in Mode 6, and at 1000 hours, CHM Technician B called the Unit Supervisor and asked about the FWST status. The Supervisor stated that the FW pump had been tagged out and that OPS was planning to clear the tagout later in the day.

CONCLUSION

This Technical Specification violation is attributed to Inappropriate Action, as a result of the individuals involved not recognizing the need to ensure an operable borated water source. The Chemistry personnel, though having contacted OPS personnel on numerous occasions to place the FWST in recirculation for sampling, did not pursue a timely resolution to the problems when continuing interferences occurred. In addition, the information discussed by CHM personnel and OPS personnel, concerning the T/S samples, was not carried out by OPS personnel in a timely manner to avoid missing a T/S sample. In the past, Chemistry personnel have understood that the boron concentrations are provided to OPS to fulfill the requirements of T/S 4.1.2.5.a.1. The requirements of 4.1.2.5.a.2 & 3 are supplied to the CRO by way of the Operator Aid Computer and as required in PT/1/A/4600/02 E, F, & G, Periodic Surveillance procedures. Therefore, Operations is responsible for the determination of OPERABILITY as stated in T/S 3.1.2.5. CHM personnel concluded that if OPS did not place the FWST in recirculation during the period of March 1 through 15, OPS must have maintained the BAT as the declared borated water source. In addition, CHM had been told by OPS personnel earlier in the outage that the BAT was the borated water source. Communication between the groups is considered a contributing cause in that it did not achieve the necessary clarity and responsiveness to avoid the T/S violation.

The inoperability of D/G 1B and the tagout of 1NV236B necessitated the inoperability of the BAT, due to loss of its boron injection flow path. This INOPERABILITY was declared based on T/S 4.1.2.1b, which requires at least once per 31 days that each valve in the flow path is in its correct position. The current Chemistry sampling schedule for FWST and for the BAT is established in CHM procedures. If this schedule is followed as stated, regardless of concerns with the "declared borated water source", the required analyses should be completed per T/S.

The CHM staff completed changes to Chemistry Management Procedure 3.4.17, on April 5, 1990, which state that if a system needs to be placed in recirculation to collect a T/S sample, OPS is to be informed at the time of the recirculation request, that, if the requested action is not taken by an appropriate time, a T/S violation will occur.

Chemistry Management Procedure 3.4.17 was also changed to include statements on FWST and BAT sampling enclosures which states that the inability to collect a T/S sample is considered the same as being Out-of-Spec. A T/S Operability Notification Sheet (Attachment 1 of Station Directive 3.1.15, Activities Affecting Station Operations) will be issued by Chemistry with a comment that the T/S sample is Out-of-Spec or unattainable.

As a result of this event, emphasis will be placed on ensuring clear communication, focusing on clear description of needed actions and clear understanding of the importance of such actions.

CHM Technician D called the CRO at 2037 hours on March 10, 1990, requesting a FWST sample. Unit 1 was in Mode 6. The CRO was asked to place the FWST on the recirculation pump so that the tank could be sampled in approximately 30 minutes. At 2045 hours, the CRO called CHM Technician D and said that the recirculation pump would not operate and asked if CHM could sample off of the FW pump. The CHM Technician explained that their sample point was on the line off of the recirculation pump. CHM Technician D completed sampling the FWST at 2130 hours.

On March 12, 1990, Unit 1 was in Mode 6. OPS had completed the Reactor cavity draining at 0500 hours. At 0600 hours, CHM Technician D inquired about the FWST sampling, and was told that the FWST was still aligned to the cavity and recirculation had not begun. Unit 1 entered Mode 5 at 1800 hours.

CHM Technician B called the CRO on March 14, 1990, with Unit 1 in Mode 5, to verify that the BAT was the declared borated water source, and that the latest FWST sample was collected and analyzed on March 10, 1990. At that time, CHM was informed of the inoperability of the BAT, due to 1NV236B being inoperable. Due to the 1B D/G being out of service, 1NV236B did not have an alternate power source available. CHM personnel were not aware of this condition. At 0900 hours, the CRO called CHM Technician B and stated that the FWST had been placed on the FW pump and should be ready for sampling by 1800 hours.

On March 15, 1990, Unit 1 was in Mode 5. At 0140 hours, the Unit Supervisor and CHM Technician E sampled the FWST off of a low point drain, 1FW14, Refueling Cavity to FW Pump Strainer Lo-Point Drain. This sample was taken to ensure that the FWST was sampled within the seven day time frame. At 0800 hours, CHM Technician B called the Unit Supervisor and asked about the BAT lineup and also asked if the transfer pumps were still tagged out. CHM Technician B discussed the conversation on March 14, 1990 with the CRO, stating that the FWST was the declared borated water source. CHM Technician B then asked the CRO how OPS could declare the source without sample results. The response was that the CRO was using the percent level for the FWST to consider it operable.

Following a review of the previous FWST and BAT sample results, the Primary CHM group determined that during the week of March 5 through 12, 1990, CHM personnel missed sampling the FWST on March 5, which violated T/S 4.1.2.5.A.1, sampling frequency of the borated water source.

On April 5, 1990, Unit 1 entered Mode 3, Hot Standby, at 0526 hours. Changes were approved for Chemistry Management Procedure 3.4.17 which incorporated notification to OPS of T/S required samples and the possibility of T/S violations if samples are not collected before an appropriate time. A requirement was established for use of a Technical Specification Operability Notification Sheet (TSONS) for samples that are Out-of-Spec or unattainable.

A search of the Operating Experience Program database for the past 24 months revealed two events, LER 414/89-018 and LER 414/89-05, that involved a missed Technical Specification sample. LER 414/89-018 was concerned with a missed sample of the Cold Leg Accumulator as a result of deficient communication. This event involved insufficient, unclear information communicated during CHM shift turnover. Also, an additional root cause was improper action; with no action taken when required because of lack of attention to detail. Corrective actions included meetings with the shift technicians to emphasize the need for effective turnover information. LER 414/89-05 involved Radiation Protection (RP) and a Turbine Building sump radiation monitor (2EMF31) sample which was not collected in a timely manner due to an inadequate sampling policy. In this event, RP procedures were changed to ensure correct, timely sample collection. This event is not considered a recurring event.

CORRECTIVE ACTION

SUBSEQUENT

1) Chemistry Management Procedure 3.4.17 was revised to include:

   a. Steps that will ensure that, if a system/component needs to be placed in recirculation or a valve needs to be manipulated in order to collect a T/S sample, OPS personnel are to be informed at the time of the recirculation or valve manipulation request, that if the system is not put in the configuration requested by an appropriate time, then a T/S violation will occur.

   b. Steps in Enclosures for Primary Chemistry sampling that direct the CHM Technicians to complete a T/S Operability Statement (TSONS) when a T/S sample is unattainable (which is considered to be the same as being Out-of-Spec). The TSONS will provide the specific information for OPS to follow-up direct actions pertaining to T/S operability.

PLANNED

1) OPS Shift personnel will be informed of the Chemistry section's April 5, 1990 procedure changes to 3.4.17.

2) Management will emphasize the accountability of all personnel to ensure clear communication and understanding of needed action and its importance. This effort will include review and (as much as practical) standardization of each group's methods and paths of communication with Operations. This effort will be discussed with Operations personnel with emphasis on their obligation to "reach into" interfacing activity areas and ensure understanding and appropriate action.

SAFETY ANALYSIS

The usable capacity of the FWST is based on the requirement for filling the refueling cavity to a depth that limits the radiation at the surface of the water to 2.5 mrem/hr during the period when a fuel assembly is transferred over the Reactor vessel flange. This function requires more water than is necessary for a post-LOCA safe shutdown.

The NV System maintains the coolant inventory in the NC System within the allowable pressurizer level range for all normal modes of operation. This sysem also contains sufficient makeup capacity to maintain the minimum required inventory in the event of minor NC leaks. Other than the centrifugal charging pumps and associated piping and valves, the NV System is not required to function during a LOCA. During a LOCA, the NV System is isolated except for the centrifugal charging pumps and the piping in the safety injection and seal injection path.

When the Reactor is subcritical, i.e., during Cold or Hot Shutdown, refueling and approach to criticality, the neutron source multiplication is continuously monitored and indicated. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to allow ample time to start a corrective action to prevent the core from becoming critical.

During the period from March 5 through 10, 1990, following the missed FWST boron sample analysis, the Unit was in Mode 6. The FWST was considered the declared or assured borated water souce. All parameters for tank volume, and solution temperature were maintained within required T/S limits. The boron concentration from the February 26 analysis was 2071 ppm, and the concentration from the March 10 analysis was 2148 ppm. It is considered that the concentration did not significantly decrease during this period based on the values for these two samples.

The health and safety of the public were unaffected by this incident.

*U.S. GPO: 1988-520-589 00070

EXHIBIT B-8

Cooper Station:
LER 298/86-034-00 (December 18, 1986)

While conducting an evaluation of fuel enrichment requirements to facilitate future extended cycle (18 month) operation, a review of the existing CNS Technical Specifications was made by the General Electric Company (GE) to determine the extent of any revisions that might be required. During the course of this Technical Specification review, an apparent violation of paragraph 5.5.B was identified. Paragraph 5.5.B states that, " . . . In addition, fuel in the storage pool shall have a U-235 loading of less than or equal to 14.5 grams of U-235 per axial centimeter of fuel assembly". However, GE advised that the barrier fuel, GE Type BP8DRB283, which had been supplied for Cycle 11, contained a U-235 loading of approximately 14.6 grams, in excess of the 14.5 grams per axial centimeter limit. Hence, storage of the new fuel in the Spent Fuel Storage Pool constituted a violation of the Technical Specifications. Upon receipt of this notification from GE on November 14, 1986, an evaluation was conducted of all fuel reloads that had been stored in the Spent Fuel Storage Pool. On November 18, 1986, the determination was made that the fuel supplied for Cycle 7 and stored in the Spent Fuel Storage Pool from February 3, 1981 to April 27, 1981 and the fuel supplied for Cycle 10, which was stored in the Spent Fuel Storage Pool from July 23, 1984 to July 17, 1985, also contained U-235 loading slightly greater than 14.5 grams per axial centimeter. At the time of these discoveries, the plant was in a shutdown condition for a refueling/major maintenance outage which had commenced on October 4, 1986.

This event is being reported in accordance with the requirements specified in 10CFR50.73(a)(2)(i) in that storage of fuel with a U-235 loading in excess of 14.5 grams per axial centimeter constitutes a violation of paragraph 5.5.B of the CNS Technical Specifications. It appears that this limitation is based upon the U-235 loading which corresponds to the nominal fuel design parameters associated with the fuel type considered in the safety analysis conducted to support backfit of the Spent Fuel Storage Pool in 1978 with high density fuel racks.

Amendment 52 to the CNS Technical Specifications, dated June 12. 1978, which provided for installation of high density fuel racks in the Spent Fuel Storage Pool, was issued by the NRC with the aforementioned 14.5 grams per axial centimeter limit. The criticality calculations which were performed to provide the technical basis for the new design racks were based upon General Electric type 8DR283 fuel assemblies. These assemblies had an average enrichment of 2.83 w/o and a nominal pellet density of 95.0% theoretical density (TD). The 150 inch fuel assembly design includes a 6 inch section of natural uranium at its top and bottom. The central 138 inches of these fuel assemblies contain an enrichment of 3.01 w/o. The 14.5 grams/centimeter value is based on this enrichment and the nominal density of 95.0%. In establishing this value, however, no consideration was given to deviations from nominal fuel assembly design parameters which are within the tolerances considered in the fuel designed and licensed by GE. These deviations from nominal parameters may result from either manufacturing tolerances or design improvements.

In addition, the fuel supplied by GE for Cycle 11 was manufactured with an upgraded pellet design incorporating a slightly higher theoretical density. As a result, the 14.5 grams per axial centimeter limit was exceeded. With respect to the fuel provided for Cycles 7 and 10, the axial limit was exceeded due to manufacturing tolerances within the approved design envelope.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Cooper Nuclear Station	0 5 0 0 0 2 9 8	8 6	— 0 3 4	— 0 0	0 3 OF 0 3

TEXT (If more space is required, use additional NRC Form 366A's) (17)

General Electric has advised that neither the pellet design change nor the manufacturing deviations, which are within prescribed tolerances, constitute a safety problem. Fuel enrichment had not changed, consequently fuel reactivity had not changed. Criticality calculations performed in 1978 to support issuance of Amendment 52 to the CNS Technical Specifications are still fully applicable to storage of fuel of the present design. Hence, the cause of the Technical Specification violation is attributed to the lack of consideration of allowable fuel design parameter tolerances in calculations performed to support the 14.5 grams per axial centimeter limit, coupled with a failure to recognize the impact of the slightly increased pellet density on the Spent Fuel Storage Pool limits.

Corrective action to be taken will consist of a review of Spent Fuel Storage Pool design for fuel loading and performance of calculations to update storage limits which are prescribed in the CNS Technical Specifications. Ensuing changes to the Technical Specifications determined to be appropriate will be transmitted to the NRC.

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Cooper Nuclear Station

DOCKET NUMBER (2): 0 5 0 0 0 2 9 8

PAGE (3): 1 OF 0 3

TITLE (4): Storage of Fuel in the Spent Fuel Storage Pool with U-235 Loading in Excess of Technical Specification Limits due to Pellet Design Changes & Manufacturer Variances

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 1	1 8	8 6	8 6	0 3 4	0 0	1 2	1 8	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME: D. L. Reeves, Jr.

TELEPHONE NUMBER: AREA CODE 4 0 2 | 8 2 5 - 3 8 1 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

As a result of an investigation performed by the General Electric Company, and further evaluation performed by CNS personnel, it was determined that new fuel stored in the Spent Fuel Storage Pool for Cycles 7, 10, and the current cycle, Cycle 11, contained a U-235 loading in excess of that allowed by Technical Specifications, paragraph 5.5.B. At the time of this discovery, a refueling/major maintenance outage was in progress.

The cause of this problem is twofold in that:

1) The fuel received for Cycle 11 incorporated pellets of a newer design with a nominal density slightly higher than previous designs.
2) The fuel received for Cycles 7 and 10, while manufactured within approved design tolerances, included pellets of a density in excess of the nominal value.

General Electric has advised that while the U-235 loading limit of 14.5 grams per axial centimeter specified by Technical Specifications was exceeded, the average fuel enrichment was unchanged and, therefore, the reactivity of the fuel had not been increased. Hence, the criticality calculations made in support of the high density fuel rack upgrade remain fully applicable.

Corrective actions to be taken will consist of a review of Spent Fuel Storage Pool design for fuel loading and further calculations to update storage limits prescribed in the CNS Technical Specifications.

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NRC Form 366 (9-83)

EXHIBIT B-9

Crystal River Unit 3:
LER 302/87-026-00 (December 1, 1987)

NRC Form 366

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): CRYSTAL RIVER UNIT 3

DOCKET NUMBER (2): 0 5 0 0 0 3 0 2

PAGE (3): 1 OF 0 1

TITLE (4): Personnel Error Results in Development of an Erroneous Fuel Move Sheet and Placing a Fuel Assembly in an Incorrect Location

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 1	0 9	8 7	8 7	0 2 6	0 0	1 2	0 1	8 7	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

- [ ] 20.402(b)
- [ ] 20.405(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [X] 50.73(a)(2)(i)
- [ ] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME: L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR

TELEPHONE NUMBER: AREA CODE 9 0 4 7 9 5 - 6 4 8 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

- [ ] YES (If yes, complete EXPECTED SUBMISSION DATE)
- [X] NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On November 9, 1987, Crystal River Unit 3 was shut down in a refueling outage. The reactor vessel was completely defueled to facilitate inspection of the core flood valves. Fuel and Control Rod assemblies were being moved in the spent fuel pools in preparation for the core reload. At 1715, while updating the control room fuel location tag board, it was noted that a new fuel assembly, with 3.851 percent U-235 enrichment had been placed in the "A" Spent Fuel Pool. The fuel racks in the "A" Spent Fuel Pool are limited to storage of fuel assemblies with 3.5 percent or less U-235 enrichment. This event was caused by a personnel error. When move sheets were being prepared to move a fuel assembly from location M42 in the "B" Spent Fuel Pool to the "A" Spent Fuel Pool, location M43 was inadvertantly written instead of M42. The mislocated fuel assembly was removed from the "A" Spent Fuel Pool upon detection of its mislocation. Independent review of move sheets, prior to actual fuel movement, has been implemented.

NRC Form 366

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO 3150-0104 EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): CRYSTAL RIVER UNIT 3

DOCKET NUMBER (2): 0 5 0 0 0 3 0 2

PAGE (3): 1 OF 0

TITLE (4): Personnel Error Results in Development of an Erroneous Fuel Move Sheet and Placing a Fuel Assembly in an Incorrect Location

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
1 1	0 9	8 7	8 7	0 2 6	0 0	1 2	0 1	8 7	N/A	0 5 0 0 0
									N/A	0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

- [ ] 20.402(b)
- [ ] 20.405(a)(1)(i)
- [ ] 20.405(a)(1)(ii)
- [ ] 20.405(a)(1)(iii)
- [ ] 20.405(a)(1)(iv)
- [ ] 20.405(a)(1)(v)
- [ ] 20.405(c)
- [ ] 50.36(c)(1)
- [ ] 50.36(c)(2)
- [X] 50.73(a)(2)(i)
- [ ] 50.73(a)(2)(ii)
- [ ] 50.73(a)(2)(iii)
- [ ] 50.73(a)(2)(iv)
- [ ] 50.73(a)(2)(v)
- [ ] 50.73(a)(2)(vii)
- [ ] 50.73(a)(2)(viii)(A)
- [ ] 50.73(a)(2)(viii)(B)
- [ ] 50.73(a)(2)(x)
- [ ] 73.71(b)
- [ ] 73.71(c)
- [ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME: L. W. MOFFATT, NUCLEAR SAFETY SUPERVISOR

TELEPHONE NUMBER: AREA CODE 9 0 4 7 9 5 - 6 4 8 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

- [ ] YES (If yes, complete EXPECTED SUBMISSION DATE)
- [X] NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On November 9, 1987, Crystal River Unit 3 was shut down in a refueling outage. The reactor vessel was completely defueled to facilitate inspection of the core flood valves. Fuel and Control Rod assemblies were being moved in the spent fuel pools in preparation for the core reload. At 1715, while updating the control room fuel location tag board, it was noted that a new fuel assembly, with 3.851 percent U-235 enrichment had been placed in the "A" Spent Fuel Pool. The fuel racks in the "A" Spent Fuel Pool are limited to storage of fuel assemblies with 3.5 percent or less U-235 enrichment. This event was caused by a personnel error. When move sheets were being prepared to move a fuel assembly from location M42 in the "B" Spent Fuel Pool to the "A" Spent Fuel Pool, location M43 was inadvertently written instead of M42. The mislocated fuel assembly was removed from the "A" Spent Fuel Pool upon detection of its mislocation. Independent review of move sheets, prior to actual fuel movement, has been implemented.

NRC Form 366 (9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)	PAGE (3)
CRYSTAL RIVER UNIT 3	0 5 0 0 0 3 0 2	87 — 026 — 00	03 OF 03

PREVIOUS SIMILAR EVENTS

This is the first occurrence of this type at Crystal River Unit 3.

EXHIBIT B-10

Hope Creek Station:
LER 354/95-042-00 (March 25, 1996)

NRC FORM 366 (4-95) — U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HR. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
HOPE CREEK GENERATING STATION	05000-354	1 OF 4

TITLE (4)

Fuel Bundle Confirmed to be Misoriented during an Operating Cycle

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
12	12	95	95	-- 042 --	00	03	25	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 5

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	X	OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A Voluntary Report
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeff Keenan, Licensing	609 - 339 - 5429

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 12, 1995, one reactor core fuel bundle was verified to be misoriented by 180 degrees. This bundle was confirmed to have been misoriented for the last cycle of operation. The event occurred during the last refueling outage (RFO5) when a refuel bridge operator failed to correctly rotate a bundle when moving it within the reactor core. In addition, the independent verification processes failed to identify the error. There was no safety consequence to plant operation due to this event; however, to share industry information this report is being submitted voluntarily.

Causes of this event are less than adequate procedural and human factor controls being established for the core verification process. Corrective actions included revisions to procedures and additional training with personnel performing core verification activities. In addition, an assessment of fuel movement practices will be completed prior to the next refueling outage.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)

IDENTIFICATION OF OCCURRENCE

TITLE: Fuel Bundle Confirmed to be Misoriented during an Operating Cycle

Event Date: December 12, 1995

CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 5 (Refueling)
Reactor at 0% of Rated Power

DESCRIPTION OF OCCURRENCE

On December 12, 1995, while shutdown for refueling, a visual inspection of the reactor core by refueling bridge personnel revealed a fuel bundle that was apparently 180 degrees out of proper orientation. Supervision was immediately notified and the bundle was verified to be misoriented. The misoriented bundle was positioned in a North-East (NE) orientation in lieu of the proper South-West (SW) orientation. A review of core verification video tapes from previous refueling outages confirmed that the bundle was misoriented during the last cycle of operation.

A review of records has revealed that the mispositioning occurred at 0736 hours on Sunday, April 3, 1994. The bundle was picked up in a NE orientation and not rotated to the SW orientation during the fuel move. Core verification, comprising a video monitor review of the core, was performed at that time. As part of the verification, bundle orientation was reviewed by looking at four bundles at a time (a fuel cell) during a continuous scan of the core by the refueling bridge camera.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
HOPE CREEK GENERATING STATION	05000-354	95	042	00	3 OF 4

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

ANALYSIS OF OCCURRENCE

Fuel assemblies are arranged in the core according to a design that meets reactivity control requirements and core operating limits. Bundle orientation is an attribute which has an effect on this design. Multiple administrative barriers are in place to decrease the probability of bundle misplacement. Bundle placements are controlled according to procedures "Conduct of Fuel Handling" (NC.NA-AP.ZZ-0049(Q)) and "Refueling Platform and ~~uel Grapple Operation" (HC.OP-SO.KE-0001(Q)). These procedures require ~~uel moves to be independently verified by the refueling floor bridge operator, spotter and refueling Senior Reactor Operator (SRO). A channel fastener (spring clip), located on top of the fuel assembly, acts as a physical aid in ensuring proper bundle orientation. In addition, after all fuel movements are completed, a core verification is performed in accordance with procedure "Verification of Fuel Location" (HC.RE-FR.ZZ-0008(Q)). This procedure specifically requires two scans of the core, one for identification numbers and the other for proper orientation. Additionally, this procedure had incorporated the recommendations of Service Information Letter (SIL) 347 concerning misoriented fuel bundles.

Any one of the above discussed barriers alone should have prevented the event. However, the fuel was misoriented by the refueling bridge operator, not accurately verified by the other bridge operating personnel, and not accurately verified during the independent core verification.

PARENT CAUSE OF THE OCCURRENCE

The causes for the initial bundle placement and fuel bridge verification errors have been inconclusive. The long time before discovery of the event has hindered the collection of relevant personnel data surrounding the events on the bridge at the time of the error. Although unable to develop a definitive causal factor, a comprehensive corrective action is in place to critically review fuel movement practices.

The procedures for core verification have been reviewed and have been determined to be deficient in detail, scope and level of independent review. Specifically, the procedure was less than adequate in providing sufficient detail for "independent" reviews. Scope of the procedure was less than adequate in that it emphasized serial number checking over orientation and was ambiguous regarding the secondary review being limited to serial numbers. In addition, the procedure had less than adequate consideration for human factors controls in the taping and verification review. Finally, there was an inadequate self verification process for documenting the orientation check and having review aids for the orientation check.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION	
HOPE CREEK GENERATING STATION	05000-354	95 --	042 --	00	4 OF 4

TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

There were less than adequate human factor controls built into the core verification process. Verifiers document the bundle number; however, for the orientation check they are reviewing the monitor passively and react only if a problem is observed. In addition, the monitor's focus tended to be only on the channel clips. A view of the complete fuel cell would allow the verifier to have multiple indicators to assess proper orientation. A strengthening of these human factors issues will further reduce the probability of a fuel bundle misorientation event.

SAFETY SIGNIFICANCE

This event had no safety significance. The misoriented fuel bundle and the adjacent fuel bundles, operated within fuel design limits during the cycle of concern. A thorough analysis concluded that thermal power, shutdown margin, average linear heat generation rate, minimum critical power ratio and linear heat generation rate were all minimally affected. Technical Specification limits were maintained throughout the cycle.

PREVIOUS OCCURRENCES

There have been no previous reported events involving a fuel bundle being misoriented for a cycle of operation. However, a limited number of fuel bundle seatings and one misorientation have been corrected during the core verification process in the past.

CORRECTIVE ACTIONS

1) The procedure for "Verification of Fuel Location", HC.RE-FR.ZZ-0008(Q), was revised prior to the current outages core verification to correct inadequacies concerning detail, scope, and self verification.

2) The event was reviewed and self verification was stressed with current fuel handlers and reactor engineers prior to recommencing fuel movement.

3) A comprehensive assessment of fuel movement practices will be performed. The assessment will be completed prior to the next refueling outage.

EXHIBIT B-11

McGuire Unit 1:
LER 369/94-005-00 (August 10, 1994)

RECEIVED

'94 AUG 29 P4:15

August 10, 1994

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: McGuire Nuclear Station Unit 1
Docket No. 50-369
Voluntary Licensee Event Report 369/94-05
Problem Investigation Process No.: 1-M94-0801

Gentlemen:

Attached is a voluntary Licensee Event Report 369/94-05 concerning the Boron dilution of the Unit 1 Spent Fuel Pool during drain down and decontamination of the Transfer Canal. This report is being submitted voluntarily and is not required per 10 CFR 50.73. This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

T.C. McMeekin

RJD/bcb

Attachment

xc: Mr. S.D. Ebneter
Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., NW, Suite 2900
Atlanta, GA 30323

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, GA 30339

Mr. Victor Nerses
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. George Maxwell
NRC Resident Inspector
McGuire Nuclear Station

IE22

bxc: B.L. Walsh (EC11C)
P.R. Herran (MG01VP)
R.C. Norcutt (MG01WC)
K.L. Crane (MG01RC)
B.F. Caldwell (MG01VP)
R.N. Casler (EC05N)
S.G. Benesole (ONS)
G.H. Savage (EC06E)
G.B. Swindlehurst (EC11-0842)
M.S. Tuckman (EC07H)
R.F. Cole (EC05N)
D.B. Cook (EC13A)
G.A. Copp (EC05O)
Tim Becker (PB02L)
J.I. Glenn (MG02ME)
P.M. Abraham (EC08I)
Zach Taylor (CNS)
L.V. Wilkie (CN03SR)
D.P. Kimball (ON05SR)
NSRB Support Staff (EC 12-A)

NRC FORM 366 (5-92)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME(1)	DOCKET NUMBER(2)	PAGE(3)
McGuire Nuclear Station, Unit 1	05000 369	1 OF 7

TITLE(4) Boron Dilution of the Unit 1 Spent Fuel Pool During Drain Down and Decontamination of the Transfer Canal.

EVENT DATE(5)			LER NUMBER(6)			REPORT DATE(7)			OTHER FACILITIES INVOLVED(8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
										05000
07	11	94	94	05	0	08	10	94		05000

OPERATING MODE(9): 1

POWER LEVEL(10): 100%

THIS REPORT IS SUBMITTED PURSUANT TO REQUIREMENTS OF 10CFR (Check one or more of the following)(11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	X	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER(12)

NAME	TELEPHONE NUMBER	
	AREA CODE	
Ricky J. Deese, Manager, McGuire Safety Review Group	704	875-4065

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT(13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED(14)				EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines (16)

This report is being submitted voluntarily to provide information and lessons learned regarding a Reactivity Management Event. On July 10, 1994, with Unit 1 operating in Mode 1 (· Operation) at 100 percent power, Mechanical Maintenance personnel began the drain down of the Unit 1 Spent Fuel Pool Transfer Canal. During the drain down, a demineralized water misting system was used to keep the pool walls wet to minimize potential airborne contamination. Approximately 28,000 gallons of demineralized water was added to the pool during the decontamination process. The addition of the demineralized water lowered the Boron concentration from 2105 parts per million (ppm) to 1957ppm. The Technical Specification requires a Boron concentration >/= 2000ppm. The Action Statement to suspend fuel movement while the Boron concentration is less than 2000ppm was not violated. Boric Acid was added to the pool to bring the Boron concentration above 2000ppm. This event has been assigned a cause of improper Managerial Methods. Corrective actions include heightening the awareness of site personnel to Reactivity Management concerns, evaluation of work processes/controls, rewrite of the procedure used, incorporation of work involving complex evolutions and multiple interfaces into the Risk Assessment Process.

LER FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 5/31/95
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION		ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME(1)	DOCKET NUMBER(2)	LER NUMBER(6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE(3)		
McGuire Nuclear Station, Unit 1	05000 369	94	05	0	2	OF	7

This is a voluntary LER.

EVALUATION:

Background

Valve [EIIS:ISV] 1KF-122, Fuel Transfer Tube Isolation, is located in the Spent Fuel Pool (SFP) Transfer Canal and is used to isolate the SFP from the Refueling Cavity in the Reactor Building. During normal operation, a blank flange is installed on the Reactor Building side of the Fuel Transfer Tube and valve 1KF-122 is open. This allows SFP water to enter the Fuel Transfer Tube supplying a source of borated water to the Standby Makeup Pump. This pump is part of the Standby Shutdown System (SSS) and provides water to the Reactor Coolant (NC) system [EIIS:AB] and the NC pump [EIIS:P] seals if normal sources are lost. The SSS is required to be operable during Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby). Technical Specification 3.9.12a requires the Boron concentration in the SFP to be maintained at >/= 2000 parts per million (ppm). The associated action statement requires that all fuel movement be suspended if the Boron concentration is found to be below 2000ppm.

Description of Event

This report is being submitted voluntarily to provide information and lessons learned regarding a Reactivity Management Event. On July 5, 1994, with Unit 1 operating in Mode 1 (Power Operation) at 100 percent power, Mechanical Maintenance personnel performed preliminary work in preparation for the drain down of the Fuel Transfer Canal (FTC). The work included the installation of approximately 26 feet of 3/4 inch PVC pipe along both sides of the FTC. Approximately 1/16 inch holes had been drilled in the pipe at 3 to 5 inch intervals. The pipe was capped at one end and connected to a standard 3/4 inch hose on the other end. The hose was connected to a demineralized water line, but not charged. The purpose of the PVC pipe was to provide a mist of water to the walls of the FTC while the canal was being drained. This would ensure that the walls stayed wet to minimize potential airborne contamination.

On July 10, 1994, at approximately 0030, Mechanical Maintenance personnel prepared to drain down the FTC to allow the Fuel Transfer Tube Isolation valve, 1KF-122 to be replaced. Prior to beginning work, the team held a pre-job briefing and contacted Operations personnel to obtain approval to begin work.

The Mechanical Maintenance Team installed the Weir Gate and inflated the seals per Operations Procedure OP/0/A/6550/14, Draining and Filling of Spent Fuel Pool Transfer Canal and Cask Area. Operations personnel tagged the valve supplying the air to the seals in the open position. The Maintenance Team then lowered a submersible pump into the FTC and contacted the supervisor of a multi-skilled shift work team (SPOC) responsible for draining the FTC. A SPOC Team member was assigned to monitor the drain down process and ·re the pump when the canal was empty. The Maintenance Team started the pump and ...ed on the mister system to keep the FTC walls wet.

The Maintenance Team instructed the SPOC Team member to monitor SFP level, Weir Gate seal pressure, and pump operation. The SPOC Team member was also asked to check the Weir Gate seals for leaks and ensure that the FTC walls stayed wet to minimize potential airborne contamination. During the day shift on July 10, 1994, Operations Control Room personnel went to the SFP Building and observed the drain down/mister operation. The Control Room Staff discussed the effects of the mister system on Boron concentration in the SFP. They referred to the SFP makeup procedure and decided that the system would not add more demineralized water to the pool than was allowed by the makeup procedure.

At approximately 2045, the drain down was complete and the pump was secured. To ensure that the FTC walls stayed wet, the mister system was allowed to continue to run. No ific instructions had been given to the SPOC team about turning it off.

On July 11, 1994, the Maintenance Team pumped the water that was added to the FTC by the mister system out of the FTC so the Mechanical Maintenance team could begin work on valve 1KF-122. They also throttled the mister system back to reduce the amount of water being added to the FTC. Radiation Protection personnel had taken radiation level readings and believed the risk of airborne contamination had been reduced.

On July 12, 1994, Radiation Protection personnel contacted Chemistry personnel and informed them about the demineralized water that had been added to the pool. There was a concern about the amount of water that had been added by the mister system and its effect on the Boron concentration in the pool. Chemistry personnel completed sampling of the pool at 1100 and determined the Boron concentration to be 1957ppm. Enough Boric Acid was added to the pool, to raise the concentration above the Technical Specification requirement of >/= 2000ppm.

LERFORM 366A U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT
(LER) TEXT CONTINUATION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

FACILITY NAME(1)	DOCKET NUMBER(2)	LER NUMBER(6)			PAGE(3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
McGuire Nuclear Station, Unit 1	05000 369	94	05	0	4	OF	7

Conclusion

This event is assigned a cause of improper Managerial Methods. The following is a list of examples/contributing factors.

1) The personnel responsible for execution support for the Maintenance Team allowed the ing system that had been used in the past to be altered without reviewing impact on demineralized water flow and thus Boron concentration.

2) The turnover of the job between the Maintenance Team and the SPOC Team was not adequate. The Maintenance Team was familiar with the procedure and was aware of the note in the procedure that stated, "The continuous use of misting hoses will add a substantial amount of water which when pumped over can cause pool dilution". They did not inform the SPOC of the note and the need to be concerned about how much water was added.

3) Operations personnel questioned the addition of demineralized water to the pool, but did not verify Boron concentration of the pool or ensure that adequate controls were in place to prevent over dilution.

4) The part of the job associated with drain down of the FTC was not discussed or planned in detail. Since the drain down was being performed by an existing procedure and had been performed before without incident, no one saw a need to review the process. The plan for the modification should have included all aspects of the job, including drain down and decontamination of the FTC.

5) Personnel involved with the actual drain down did not see the note in the procedure concerning the potential for diluting the pool and did not recognize that the mister system could significantly affect the Boron concentration of the pool. Personnel interviewed did not have a good understanding of their responsibilities associated with Reactivity Management (Nuclear System Directive 304).

6) The incorrect tags were hung on the air supply valves for the Weir Seals. OP/0/A/6550/14 specifies red tags (Employee Safety) to be hung on the valves. Operations personnel hung white tags (Equipment Safety) on the valves. The procedure was not followed as required.

7) The SPOC team was not qualified to the procedure and had not run the procedure previously. This situation requires that the Supervisor or qualified individual give close direction to the employees involved to ensure adequate completion of the task assigned.

8) The decision, on July 11, to pump the additional water out of the FTC, without rmining the full impact was in error. Emphasis was on the work schedule and desire to ..urn the SSS to operation as soon as possible. The Job Sponsor, Radiation Protection Technician, Mechanical Maintenance Valve Supervisor, Work Window Manager, Maintenance Team Members, and the Maintenance Team Support Technician, reviewed the situation; however, the amount of demineralized water in the FTC was unknown. The possibility that this amount of water could lower the Boron concentration of the SFP below 2000ppm was not considered.

Corrective actions to prevent recurrence include heightening the awareness of site personnel to Reactivity Management concerns, evaluation of work processes/controls, rewrite of procedure OP/0/A/6550/14 to better clarify the concern for ensuring the misting system does not add enough water to effect SFP Boron concentration, and incorporation of work invoiving complex evolutions and multiple interfaces into the Risk Assessment Process.

view of the Problem Investigation Process data bases for the past 24 months revealed .. event related to Reactivity Management. Therefore, this event is not considered to be recurring.

This event is not Nuclear Plant Reliability Program (NPRDS) reportable.

There were no radiation overexposures, or uncontrolled releases of radioactive material resulting from this event.

CORRECTIVE ACTIONS:

Immediate: 1) Chemistry personnel added approximately 1000Kg of Boric Acid to the pool.

2) Mechanical Maintenance personnel isolated the Mister system and only used it intermittently to wet the walls.

Subsequent: Site Management has clarified that the Nuclear Engineering Group is responsible for work associated with the Spent Fuel Pool until improved processes/controls are in place.

Planned:

1) Nuclear Engineering personnel will identify and implement a method to heighten the awareness by appropriate site personnel to Reactivity Management concerns.

2) Nuclear Engineering will evaluate work associated with the Spent Fuel Pool and recommend improved processes/controls to ensure concerns such as Foreign Material Exclusion, Dilution, Fuel integrity etc. are properly addressed.

3) Maintenance Procedure Group will coordinate with Operations and Nuclear Engineering to rewrite OP/0/A/6550/14 to specifically address the decontamination activities.

4) Superintendent of Mechanical Maintenance will ensure that the Risk Assessment process includes a review of work involving complex evolutions and multiple interfaces, not covered by existing processes, to determine if Project Managers are needed.

5) Safety Assurance personnel will lead a review of the Work Control process using the problems identified in this event as examples of specific areas to address.

SAFETY ANALYSIS:

This event had no safety significance and is being provided voluntarily to provide information and lessons learned regarding a Reactivity Management event. The Spent Fuel Pool is designed to contain borated water at >/= 2000ppm Boron. However, the Licensing Basis for the plant does not take any credit for dissolved Boron in the pool for normal operation. The borated water in the pool serves two purposes. One purpose is to provide an additional margin of reactivity control above that which is required by the Final

Safety Analysis Report. It also serves as a source of borated water for the Standby Makeup pump.

The Standby Makeup pump was removed from service to allow draining of the FTC. Therefore, the possibility of the diluted water being pumped into the NC System was eliminated. A[illegible], the effect on reactivity control within the pool was minimal. Boron concentration [illegible] only two and one half percent below the Technical Specification limit. The Licensing [illegible] for the plant takes no credit for dissolved Boron in the pool under normal conditions. The fuel storage racks provide all of the negative reactivity required to keep K(eff) below .95.

The Technical Specification Action Statement requires that all fuel movement be suspended, if the Boron concentration in the pool drops below 2000ppm. No nuclear fuel was moved; therefore, at no time during this event was the Technical Specification Action Statement violated.

At no time were the health and safety of the public or plant personnel affected by this event.

EXHIBIT B-12

McGuire Unit 1:
LER 369/91-0160-00 (November 25, 1991)

LICENSEE EVENT REPORT (LER)

FACILITY NAME(1)	DOCKET NUMBER(2)	PAGE(3)
McGuire Nuclear Station, Unit 1	05000 369	1 OF 5

TITLE(4) Qualified Fuel Assemblies Were Stored Improperly In The Unit 1 Spent Fuel Pool Due to A Defective Procedure.

EVENT DATE(5)			LER NUMBER(6)			REPORT DATE(7)			OTHER FACILITIES INVOLVED(8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
10	24	91	91	16	0	11	25	91	N/A	05000
										05000

OPERATING MODE(9): NM

POWER LEVEL(10): 0%

THIS REPORT IS SUBMITTED PURSUANT TO REQUIREMENTS OF 10CFR (Check one or more of the following)(11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER (Specify in Abstract below and in Text)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER(12)

NAME	TELEPHONE NUMBER AREA CODE	
.rry L. Pedersen, Supervisor, Safety Review Group	704	875-4487

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT(13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED(14)		EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines (16)

While reviewing Technical Specification Section 3.9.12, McGuire Reactor Unit personnel identified 11 fuel assemblies that had been stored in the Unit 1 Spent Fuel Pool in a manner contrary to the requirements of Technical Specification 3.9.12. This Limiting Condition For Operation requires, in part, that fuel stored in Region 2 of the Spent Fuel Pool shall undergo 16 days of decay, and if a checkerboard pattern is employed for unqualified fuel, one row between normal storage locations and checkerboard storage locations will be vacant. The vacant row provision of the specification was not satisfied from March 23, 1990 through October 23, 1991. At the time of discovery at 0900 on October 24, 1991, Unit 1 was defueled, and Unit 2 was in Mode 1 (Power Operation) at 100 percent power. This event has been assigned a cause of Defective Procedure. The fuel assemblies in question were immediately moved to positions to establish the required vacant row.

EVALUATION:

Background

The Unit 1 Spent Fuel Pool (SFP) is composed of two regions of high density storage racks [EIIS:RK]. Region 1, which contains 286 locations, has a high density fuel assembly spacing of 10.4 inches on center. This spacing is obtained by using a neutron absorbing material. Region 1 is reserved for temporary core off loading of spent fuel assemblies. Region 2, which contains 1177 locations, has a high density fuel assembly spacing of 9.125 inches on center. Region 2 provides normal storage for irradiated fuel assemblies.

Technical Specification (TS) 3.9.12 states that unrestricted storage of spent fuel, in Region 2, shall be limited to fuel assemblies of a specified burnup within the acceptable range of TS Table 3.9-1, Minimum Burnup Versus Initial Enrichment for Region 2 Storage. Additionally, the TS requires that fuel not meeting the burnup criteria specified in TS Table 3.9-1 must be stored in a checkerboard fashion (empty locations on each side of the spent fuel assembly) with an open row between the checkerboard and normal storage locations if stored in Region 2.

Free standing fuel assembly inserts, dummy assemblies, fuel storage racks and fuel assemblies are transferred within the same unit using procedure OP/O/A/6550/11, Internal Transfer of Fuel Assemblies. Steps 3.1 through 3.6 of the procedure detail the process employed by the Reactor Unit (RU) Engineers in determining the fuel assembly storage locations. Enclosures 4.1, Internal Transfer Data Sheet and 4.4, Verification of Assemblies to be placed in Region 2, document the assembles initial and final locations, transfer dates, and required reviews and approvals.

Description of Event

On March 13, 1990, RU Engineer A completed Enclosures 4.1 and 4.4 as directed by step 3.1.1 of procedure OP/O/A/6550/11. RU Engineer A forwarded the enclosures to RU Engineer B for review and approval.

On March 23, 1990, nine of the eleven previously designated and approved final fuel assembly locations were changed by RU Engineer A at the request of the Operations Fuel Handling Supervisor to maximize available storage cells in preparation for the next core off load scheduled during Unit 1 End of Cycle (EOC) 7. Procedure OP/O/A/6550/11 does not specifically address the necessity of generating a new Enclosure 4.1 or 4.4 when final locations are revised. Consequently, the locations for 9 of the 11 qualified fuel assemblies originally recorded on Enclosure 4.1 on March 13, 1990 were deleted by line-through and the new locations were entered on the enclosure. Enclosure 4.1 was forwarded to the Maintenance Fuel Handling crew who transferred the assemblies in question to locations specified by RU Engineer A. The records indicate that the assemblies remained in these locations until the event discovery date on October 24, 1991.

Conclusion

This event has been assigned a cause of Defective Procedure due to a Technical Deficency because the procedural guidance provided by procedure OP/O/A/6550/11 is obscure. The procedure attempts to convey the intent of TS 3.9.12, but the phrasing of the procedure, especially Enclosure 4.4, leads the individual completing the procedure in a direction that does not comply with the full requirements of TS 3.9.12. For example, Enclosure 4.4 states: "Verify all fuel assemblies to be placed in Region 2 of the Spent Fuel Pool are within the limits of Technical Specification 3.9.12 and Enclosure 4.5 (see Step 2.3) by checking the assemblies' design and burnup documentation". This leads one to believe that by checking the design and burnup documentation, the TS and Enclosure 4.5 requirements will be satisfied. This is not the case. Also, although the "checkerboard pattern" is referred to in the procedure, the only reference to the open row requirement is contained in the section of the TS Limiting Condition for Operation (3.9.12.b(3)) pertaining to the storage of unqualified fuel. The storage of unqualified fuel is governed by the requirements of procedure OP/O/A/6550/11 and TS 3.9.12, i.e. checkerboard array and physical barriers. These requirements would prevent the violation of the open row provision with unqualified fuel. The mis-storage of qualified fuel assemblies would be the most probable method of violating the open row. Therefore, to enhance clarity and accuracy, procedure OP/O/A/6550/11 and TS 3.9.12 should address the open row requirement and its association with the storage of qualified versus unqualified fuel. Additionally, the TS requirements are not fully included in procedure OP/O/A/6550/11. This requires the individual performing the procedure and the procedure reviewer to either stop work on the procedure to retrieve the information from TS or to rely on memory to verify that all TS requirements have been satisfied. This is an undesirable situation since the procedure should be a "stand alone" tool and contain all information necessary to successfully complete the task.

This event is not Nuclear Plant Reliability Data System reportable.

A review of the Operating Experience Program Database for 24 months prior to this event identified three LERs, 369/90-14, 369/90-10, and 369/90-33 that were assigned a cause of Defective Procedure due to a Technical Deficiency. None of these LERs involve the same equipment or groups, therefore, this event is not recurring.

There were no personnel injuries, radiation overexposures, or uncontrolled releases of radioactive material as a result of this event.

CORRECTIVE ACTIONS:

Immediate: 1) RU personnel determined the cell locations necessary to re-establish the vacant row.

2) Maintenance Fuel Handling personnel moved the fuel assemblies in question to new cell locations determined by the RU personnel.

Planned: 1) Procedure OP/O/A/6550/11, Internal Transfer of Fuel Assemblies, will be revised by RU personnel to address all TS 3.9.12 requirements, specifically maintenance of the vacant row provision, and to require the completion of additional copies of Enclosures 4.1 and 4.4 as necessary to document changes in fuel assembly locations.

2) RU personnel will review and revise as necessary other procedures involving fuel movement to ensure that the procedures have adequately addressed all acceptance criteria.

3) RU Training personnel will initiate additional training associated with Reactivity Management.

SAFETY ANALYSIS:

TS 3/4.9.12.b (3) requires unqualified fuel to be stored in a checkerboard configuration in the Spent Fuel Storage Pool. In the event checkerboard storage is used, one row between normal storage locations and checkerboard storage locations is to be kept vacant.

General Office Nuclear Engineering (NE) personnel have evaluated the impact on criticality safety caused by the noncompliant fuel pool geometry. Using the Keno Va module in the SCALE III system of computer [EIIS:CPU] codes, NE has determined that the loss of the vacant row between the checkerboarded and normal storage regions does not increase the Spent Fuel Pool K eff beyond the value reported in the licensing basis. Therefore, Reactivity Management has not been jeopardized by placing the assemblies in the vacant row. Additionally, the Boron concentration in the pool, which was not considered in the above analysis, has been maintained at >/= 2000ppm and contributes an extra margin of safety. Therefore, unexpected criticality resulting from the mispositioned fuel assemblies is not a concern.

This event did not affect the health and safety of the public.

EXHIBIT B-13

Millstone Unit 2:
LER 336/92-003-01 (June 25, 1992)

NRC Form 366 (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	1 OF 0 4

TITLE (4) Spent Fuel Pool Criticality Analysis Error

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 2	1 4	9 2	9 2	— 0 0 3 —	0 1	0 6	2 5	9 2		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 3 0

THIS REPORT IS BEING SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.402(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73.(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[X] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Robert A. Borchert, Unit 2 Reactor Engineer, Ext. 4418	AREA CODE 2 0 3 4 4 7 - 1 7 9 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	D B	R K	C 4 9 0	N					

SUPPLEMENTAL REPORT EXPECTED (14)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 14, 1992, at 1415 hours, with the plant in Mode 1 at 30% power, Northeast Nuclear Energy Company (NNECO) was notified by ABB-Combustion Engineering (ABB-CE) that a calculational error existed in the criticality analysis for the Region 1 spent fuel storage racks. NNECO determined that this condition was reportable as a condition outside of the design basis of the plant. An immediate report was made to the NRC, and the existing reactivity condition of the spent fuel pool was verified to be in compliance with the plant Technical Specifications.

The original effective multiplication factor (K_{eff}) calculated by ABB-CE for the Region 1 fuel storage racks for nominal dimensions, nominal spent fuel pool temperature and 4.5 weight percent enriched fuel assemblies was 0.9224 (without uncertainties). The discovered error results in an underprediction of approximately 0.04 delta K_{eff}. Revised calculations by ABB-CE indicate that K_{eff} is actually 0.963 for the same conditions. An investigation by ABB-CE has traced the error to two approximations used in their calculation.

Criticality analyses to support spent fuel storage rack design changes are complete, and proposed changes to the plant Technical Specifications were submitted to the NRC on April 16, 1992. These changes were approved by the NRC on June 4, 1992.

TEXT (If more space is required, use additional NRC Form 366A's) (17)

I. Description of Event

On February 10, 1992, at approximately 1130 hours, Northeast Utilities (NU) was notified by an independent contractor that a higher than expected effective multiplication factor (K_{eff}) was calculated for the Region 1 fuel storage racks. On February 11, 1992, NU notified ABB-Combustion Engineering (ABB-CE) of the potential error in the spent fuel pool criticality analysis. On February 14, 1992, at 1415 hours, with the plant in Mode 1 at 30% power, Northeast Nuclear Energy Company (NNECO) was notified by ABB-CE that a calculational error existed in the criticality analysis for the Region 1 spent fuel storage racks.

The Millstone 2 spent fuel storage racks were modified in May 1986, and consist of two regions:

(a) Region 1 is designed to store up to 384 fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235. Region 1 was designed to allow fuel assembly storage in every location. The Region 1 storage racks contain a neutron poison material (Boroflex), and have a nominal center-to-center pitch of 9.8 inches.

(b) Region 2 is designed to store up to 728 fuel assemblies which have sustained at least 85% of their design burnup. Fuel assemblies are stored in a three-out-of-four array, with blocking devices installed to prevent inadvertent placement of a fuel assembly in the fourth location. The Region 2 storage racks have a nominal center-to-center pitch of 9 inches.

The original effective multiplication factor (K_{eff}) calculated by ABB-CE for the Region 1 fuel storage racks for nominal dimensions, nominal spent fuel pool temperature and 4.5 w/o enriched fuel assemblies is 0.9224 (without uncertainties). The discovered error results in an underprediction of approximately 0.04 delta K_{eff}. Revised calculations by ABB-CE indicate that K_{eff} is actually 0.963 for the same conditions. Evaluations by ABB-CE have confirmed that the Region 2 fuel storage racks are not affected by the error.

NNECO determined that this condition was reportable as a condition outside of the design basis of the plant. An immediate report was made to the NRC, and the existing reactivity condition of the spent fuel pool was verified to be in compliance with the plant Technical Specifications. All fuel movement in the spent fuel pool had previously been restricted due to the observed degradation of the neutron poison material in the Region 1 fuel storage racks. No automatic or manual safety systems were required to respond to this event.

II. Cause of Event

An investigation by ABB-CE has traced the error to two approximations used in their calculation.

First, ABB-CE used an incorrect treatment of the self-shielding effect in Boraflex for the epithermal energy group. This resulted in an overestimation of the neutron absorption in Region 1 and thus a lower calculated K_{eff}.

Second, ABB-CE used a geometric buckling term corresponding to a sparsely populated and unpoisoned array as an approximation of buckling in the poisoned configuration. This approximation also contributed to a lower calculated K_{eff} in Region 1.

III. Analysis of Event

This event is being reported in accordance with 10CFR50.73(a)(2)(ii)(B), which requires the reporting of any event or condition that results in the nuclear power plant being in a condition outside the design basis of the plant.

NRC Form 366A (6-89) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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Millstone Nuclear Power Station Unit 2	0 5 0 0 0 3 3 6	9 2	— 0 0 3	— 0 1	0 3 OF 0 4

TEXT (If more space is required, use additional NRC Form 366A's) (17)

The safety consequence of this event is a potential uncontrolled criticality event in the spent fuel pool. Upon consideration of the following factors, a significant margin to a critical condition was always maintained and, therefore, the safety consequences of this event were minimal:

(a) The boron concentration of the spent fuel pool is procedurally controlled at greater than 1720 ppm, and is typically maintained at greater than 2000 ppm.

(b) All new fuel assemblies previously stored in the Region 1 fuel storage racks had been arranged in a 2 out of 4 checkerboard array.

(c) The maximum initial enrichment of any fuel assemblies previously stored in the Region 1 fuel storage racks was less than 4 weight percent U-235, which is less than the design enrichment of 4.5 weight percent U-235.

(d) All discharged fuel assemblies previously stored in the Region 1 fuel storage racks have sustained at least one cycle of burnup.

IV. Corrective Action

Criticality analyses to support spent fuel storage rack design changes are complete, and proposed changes to the plant Technical Specifications were submitted to the NRC on April 16, 1992. These changes were approved by the NRC on June 4, 1992. These changes split Region 1 into 2 regions, Region A and Region B. Region A can store up to 224 fuel assemblies, which will be qualified for storage by verification of adequate average assembly burnup versus fuel assembly initial enrichment (reactivity credit for burnup). Region B can store up to 120 fuel assemblies with an initial enrichment of up to 4.5 weight percent U-235 and other assemblies which do not satisfy the burnup versus initial enrichment requirements of either Region A or Region C (formerly Region 2). Fuel assemblies will be stored in a 3 out of 4 array in Region B, with blocking devices installed to prevent inadvertent placement or storage of a fuel assembly in the fourth location. Region C is the new designation for the existing Region 2 storage racks. This alphabetic storage rack designation is a human factors consideration, designed to minimize the probability of a fuel assembly movement error and to provide a historical distinction between the various fuel pool configuration records. The attached figure shows the new arrangement of the spent fuel pool.

V. Additional Information

There were no failed components during this event.

Similar LERs: 77-23, 80-05, 83-07, 85-01, 86-10 and 91-10

Spent Fuel Storage Racks

Manufacturer: Combustion Engineering

Model: Hi-Cap Spent Fuel Storage Module

EIIS Code: DB-RK-C490

NRC Form 366A (6-89)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92

Estimated burden per response to comply with this information collection request: 50.0 hrs. Forward comments regarding burden estimate to the Records and Reports Management Branch (p-530), U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503.

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TEXT (If more space is required, use additional NRC Form 366A's) (17)

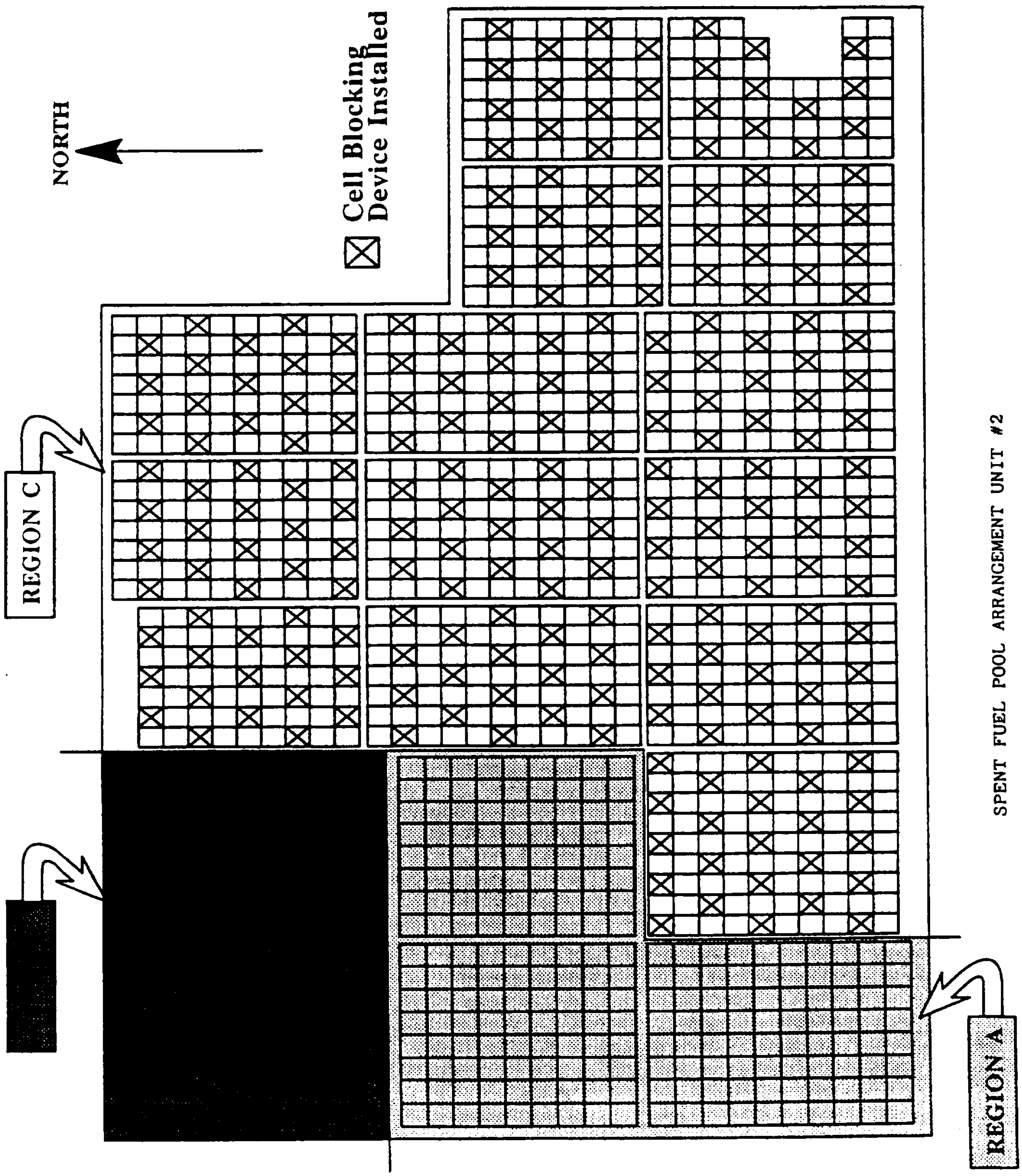

EXHIBIT B-14

Oconee Unit 1:
LER 269/96-001-00 (February 7, 1996)

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1): Oconee Nuclear Station, Unit One

DOCKET NUMBER (2): 05000 269

TITLE (4): Mispositioned Fuel Assembly Due To Inadequate Self Checking and Management Direction

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Oconee, Unit Two	05000 270
01	08	96	96	01	00	02	07	96	Oconee, Unit Three	05000 287

OPERATING MODE (9): N

POWER LEVEL (10): 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i) (B)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii) (A)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME: L. V. Wilkie, Safety Review Manager

TELEPHONE NUMBER (Include Area Code): (803) 885-3518

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO	EXPECTED SUBMISSION	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On December 14, 1995, with all three Oconee units at 100% Full power, a fuel handling team performing a fuel assembly (FA) inspection in the Unit 1&2 spent fuel pool (SFP) inadvertently left the FA unattended and suspended inside the SFP mast. It was discovered on January 8, 1996, by fuel handling personnel during check outs for planned fuel movements. The FA was reinserted into the SFP rack. The primary safety significance of the event was the potential uncovering of the FA during a postulated event requiring actuation of the Reactor Coolant Make-up function of the Standby Shutdown Facility (SSF), which uses the SFP as a water source. An engineering analysis concluded that the fuel cladding would not be breached during an SSF event with this FA in the mast. Therefore, 10CFR100 limits would not have been exceeded and the Final Safety Analysis Report (FSAR) analysis consequences would have bounded the event. However, having an unattended FA in the mast is outside the intent of Technical Specification 3.8 on fuel handling and 3.18 on the SSF. The root causes are inadequate self checking and lack of management expectations for formality and procedure use in fuel handling. Corrective actions include policy and procedure changes.

BACKGROUND

In addition to a Spent Fuel Pool (SFP) [EIIS:ND] where spent fuel is stored in racks submerged under borated water, Oconee Nuclear Station has an Interim Spent Fuel Storage Facility on site. There spent fuel is stored in dry containers, thus the term "dry cask storage" is used.

Fuel handling activities at Oconee are performed by members of a dedicated fuel handling maintenance crew. The fuel handling supervisor is a previously licensed Senior Reactor Operator. The crew's work activities are primarily fuel handling activities and plant crane [EIIS:RN] maintenance. A significant portion of the fuel handling crew's scheduled work involves shuffling spent fuel assemblies in the SFP and support of dry cask storage activities. The minimum crew number for operating the refueling bridge [EIIS:FMB] in the SFP is one bridge operator and one spotter. Fuel Handlers are qualified to Fuel Handling activities per Employee Training Qualification Standards.

OP/0/A/1506/01 (Fuel & Component Handling) is the "HOW TO" procedure for using the fuel handling bridge. It is an "Information Use" procedure which has no sign-offs, is performed from memory, and, by management policy, is not required to be at the job location.

Normally, OP/0/A/1503/09 (Documentation of Fuel Assemblies &/or Component Shuffle Within a SF Pool) is the "WHERE TO" procedure used to make miscellaneous fuel movements. An enclosure, initiated by Reactor Engineering, designates the fuel assemblies and/or control components to be moved, the starting locations, and the ending locations. The fuel handlers sign off each move as it is made.

Technical Specification 3.8 provides required prerequisites for fuel handling in the SFP. One requirement is that the SFP filtered ventilation system [EIIS:VF] must be operable, or fuel handling must be suspended. The SFP filtered ventilation system is considered inoperable whenever the fuel receiving bay door is open.

The Standby Shutdown Facility (SSF) [EIIS:NB] is designed to maintain the plant in a safe shutdown condition for a 72 hour period in the event of an Appendix R fire, a turbine building flood, a security event, a station blackout when the turbine driven emergency feedwater [EIIS:BA] pump [EIIS:P] is inoperable, or a tornado which renders the auxiliary service water and emergency feedwater systems inoperable. The SSF Reactor Coolant (RC) makeup pump [EIIS:CB] takes water from the SFP inventory in order to makeup to the Reactor Coolant System (RCS) [EIIS:AB] through the reactor coolant pump seals. In addition, SFP cooling may also be lost during an

SSF event such that boil-off of SFP water will also contribute to the loss of SFP inventory. The design basis of the SSF system will allow depletion of the SFP inventory to within one foot from the top of the SFP racks assuming no action to refill the SFP. Technical Specification 3.18.4 requires the SSF RC Makeup System be operable for each unit when the RCS is at or above 250°F.

EVENT DESCRIPTION

During Unit 1 EOC16 (End of Cycle 16) refueling outage, which started on Nov. 2, 1995 and concluded Dec 10, 1995, a fuel assembly (FA) was observed to have four intermediate spacer grids damaged. As part of the root cause evaluation, Reactor Engineer A desired to perform a visual inspection of FA NJ05T8 (FA-8), the fuel assembly which had been adjacent to the damaged assembly in the reactor core for the fuel cycle.

On December 14, 1995, at about 0900 hours, Reactor Engineer A contacted the Fuel Handling Supervisor for support in inspecting FA-8. The request was initially denied due to workload. Subsequently, one of the planned tasks was deferred several hours and the Fuel Handling Supervisor contacted Reactor Engineer A to schedule the inspection for after lunch.

Around 1300 hours, two Fuel Handlers and Reactor Engineer A entered Unit 1&2 Spent Fuel Pool (SFP) to inspect FA-8. A pre-job briefing was performed between Reactor Engineer A and Fuel Handler A but it covered only the basics of what needed to be done. Reactor Engineer A had no procedure or movement enclosure for this evolution, and, since the inspection did not involve leaving an FA in a new SFP location, Reactor Engineer A felt that he did not need one.

Fuel Handler A thought Reactor Engineer A had a procedure since he had called the control room to verify prerequisites listed in the normal fuel handling procedures. Reactor Engineer A stated that he called the control room out of habit. However, Reactor Engineer A stated that he did not inform the control room operator that fuel handling activities were about to take place.

Fuel Handler A operated the Unit 1&2 SFP bridge by memory, which is the normal practice. Fuel Handler A stated that he felt comfortable doing fuel handling steps by memory. Fuel Handler B acted as a runner for the job. Reactor Engineer A acted as a spotter, operated the video equipment, directed Fuel Handler A to SFP rack location K40, and directed mast operation (up/down) while video taping was in progress. During this

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evolution, the mast and FA-8 were moved several feet east to improve the available lighting. Also, Reactor Engineer A requested Fuel Handler A to rotate the fuel mast 90 degrees and back while FA-8 extended below the mast. After some scratches were noted on FA-8's lower end fitting, FA-8 was returned to its proper location and lowered into the storage rack.

For comparison, Reactor Engineer A decided to look at another FA selected at random from the same cycle. Reactor Engineer A directed Fuel Handler A to SFP rack location L44 to pickup FA NJ06E7 (FA-7) and directed mast operation (up/down) while the FA was video taped. After observing similar scratches on FA-7, Reactor Engineer A stated that he had seen enough.

At this point neither Reactor Engineer A nor Fuel Handler A specifically stated a need to lower FA-7 prior to proceeding.

OP/0/A/1506/01, Limit and Precaution 2.27 directs personnel to not leave portable underwater lights and cameras in close proximity to irradiated fuel assemblies when not being used. Therefore, Reactor Engineer A began to raise the video camera. Due to the need to wipe down the pole and cable attached to the camera as it is raised, this task requires two people.

However, OP/0/A/1506/01, Limit and Precaution 2.22 directs personnel to turn off the Bridge hydraulic pump to prevent overheating when a Bridge is idle for 15 minutes or greater and the hoist is not engaged. In this case the hoist was engaged, but during the investigation it was learned that the Fuel Handling Supervisor has issued standing directions to turn off the pump even if the hoist is engaged. When the hydraulic pump is off, most of the control panel indications are either de-energized or go to a default state.

In accordance with these instructions, Fuel Handler A stopped the hydraulic pump, left the control console, and assisted Reactor Engineer A with pulling up and wiping down the video equipment. Once the camera was secured, Fuel Handler A returned to the control console and de-energized the bridge. During interviews, Fuel Handler A stated that he believed that he had lowered the FA back into the fuel rack and did not look at the control console indications to confirm this.

At 1342 hours, Fuel Handlers A and B exited the Unit 1&2 SFP with Reactor Engineer A. This left FA-7 suspended and unattended in the mast.

No fuel handling tasks in the Unit 1&2 SFP occurred over the next several weeks.

On January 8, 1996, at approximately 1030 hours, Fuel Handlers A and C entered the Unit 1&2 SFP to start preparation for loading a dry cask later in the week. When Fuel Handler A energized the bridge and started the hydraulic pump, he observed the control console indications and realized that a FA was in the mast. Fuel Handlers A and C initially assumed that the FA had been left in the mast recently by other members of the crew. Fuel Handlers A and C made the decision to lower the FA in the open rack at location L44 to allow an identification of the FA in order to determine where it should be and to trace the last known movement to determine who was responsible.

While Fuel Handler A lowered FA-7 into the storage rack, Fuel Handler C called the Fuel Handling Supervisor and informed him of the discovery and that Fuel Handler A had lowered the FA in the empty rack at L44. Fuel Handlers A and C identified the FA as NJ06E7 at Unit 1&2 SFP rack location L44.

At 1130 hours, the Fuel Handling Supervisor called the Rotating Equipment Manager and Reactor Engineering to report the event. It was verified that FA-7 was the last FA moved in Unit 1&2 SFP.

At 1230 hours, a meeting was held to discuss the event. The video tape from 12/14/95 was reviewed to see if the tape had shown the FA being put back down in the pool. The personnel present concluded that FA-7 had been in the fuel mast from 12/14/95 until 1/8/96. All three Oconee units were at 100 % full power throughout this period.

The design basis of the SSF system will allow depletion of the SFP inventory to within one foot from the top of the SFP racks assuming no action to refill the SFP. A concern was raised that FA-7 could have been uncovered by an SSF event, with the potential for heating to clad failure with resultant release of fission products. However, no analysis existed to determine if clad failure would occur or if the severity of the releases would exceed limits from e[illegible]er the FSAR analysis or 10CFR100. Thus there was a concern that the SSF [illegible]ht have been unable to perform its intended function and would need to be considered past inoperable. Therefore, one action item from the 1230 meeting was to start an operability evaluation which would include calculation of expected clad temperatures and potential releases.

The Maintenance Superintendent (who was acting as the Station Manager) discussed the event during the Station Manager's staff meeting at 1330 hours. The Operations Superintendent was at the meeting and assumed the control room knew of the event.

At 1500 hours, after the staff meeting, the Maintenance Superintendent, the Rotating Equipment Manager and Fuel Handling Supervisor went to inform the ONS NRC Resident Inspectors of the event.

After briefing the senior resident, the Maintenance Superintendent, Rotating Equipment Manager, and Fuel Handling Supervisor discussed the situation and decided not to continue with fuel handling until procedures were revised to prevent this event from reoccurring.

At about 1800 hours, the Senior VP of Nuclear Generation and the Site VP discussed the event and decided to initiate a Significant Event Investigation Team (SEIT).

Throughout this period, the control room was not informed of the discovery of the FA in the mast. On 1/9/96, at about 0630 hours, an NRC resident asked control room operators about the log entry for the event. This was the first time Operations shift had heard about the event.

At 0800 hours, this event was discussed in the daily site direction meeting. Site management present discussed issues related to past operability and reportability. The information available at that time was insufficient to reach a conclusion.

At 1414 hours, a log entry was made in Unit 1 Log about the event. Notes were added on Reactor Operator (RO), Control Room Senior Reactor Operator (SRO), and Unit Shift Supervisor's turnover sheets not to move fuel in 1&2 and/or 3 SFP until after the SEIT investigation was completed.

Discussions of operability and reportability issues continued. Issues discussed included compliance with Technical Specifications (TS) and FSAR analyses of fuel damage and resultant releases. TS that potentially apply in this case are 3.8, Fuel Movement and Storage in the Spent Fuel Pool, and 3.18, Standby Shutdown Facility.

TS 3.8 was initially not considered to apply, based on an interpretation that FA-7 was not moving while left in the mast. By that interpretation, fuel handling was not in progress and, therefore, the TS was not exceeded.

TS 3.18.4 requires the SSF RC Makeup System be operable for each unit at or above 250°F in the RCS. During an SSF event the SSF RC makeup pump takes suction from the SFP and can allow depletion of the SFP inventory such that FA-7 would be uncovered. Preliminary engineering calculations indicated possible heating to clad failure with resultant release of fission products. This could result in dose consequences beyond the licensing basis.

At 1700 hours, a decision was made to make a 1 hour NRC Emergency Notification System call, based on management's conclusion that these consequences represented an unanalyzed condition that could significantly compromise plant safety. The notification was made at 1755 hours.

On 1/10/96, the SEIT arrived on site and began an investigation. On 1/12/96, the SEIT presented its preliminary findings in a formal exit with site management and the Senior VP of Nuclear Generation.

One concern raised by the SEIT was the interpretation that leaving a FA in the mast met the requirement to suspend fuel handling. A survey of industry practices revealed that all of the other sites contacted defined fuel handling to include any time an assembly was supported by the fuel handling bridge or crane. These other sites interpreted "suspension of fuel movement" to mean that fuel movement should be continued until any FA in a raised position could be moved to a safe location and lowered.

Applying this more conservative interpretation of "fuel handling" resulted in the conclusion that TS 3.8 should be applied the entire time FA-7 was in the fuel mast. Since the fuel receiving bay door was opened at various times during the period, making the filtered ventilation system inoperable, the new interpretation would mean that the intent of TS 3.7.12 was not met.

The operability calculations and analysis were completed and the results are discussed in more detail in the "Safety Analysis" section of this report. The analysis showed that FA-7 would not be damaged and would not result in off site releases exceeding 10CFR100 limits. However, another FA with a higher decay heat potentially could. Therefore, management concluded that the condition of a FA being located within the SFP mast during an SSF event is not in compliance with the intent of TS 3.18.

Therefore, in addition to being reportable as an unanalyzed condition that could significantly compromise plant safety, this event would also be reportable as a condition outside the intent of Technical Specifications.

In response to the SEIT preliminary concerns, "Short Term" actions were initiated to enhance programs, policies, and procedures to address the SEIT recommendations and observations. These were primarily aimed at those items needed to resume limited fuel shuffles in preparation for dry cask storage and new fuel receipt prior to a refueling outage on Unit 2, currently scheduled for late March, 1996.

On Feb. 1, 1996, the SEIT issued its final report. The root causes identified are the same as the root causes listed below.

CONCLUSIONS

The root causes of this event are related to inadequate barriers intended to minimize the potential for this type of error. Two root causes for the event have been determined:

The first root cause of this event is the failure of Fuel Handler A to self-check his actions. This was a skill based error resulting from a momentary memory lapse while performing routine actions using an Information Use procedure.

The second root cause to this event is the lack of management expectations for formality in all aspects of the fuel handling process. The lack of formality was exhibited in the following actions, which were in accordance with management's expectations at the time for this type of work in the spent fuel pool, leading up to the leaving of the FA in the mast:

1. The failure to write and process a work request for the conduct of this activity.

2. The perception that no task specific procedure was required to conduct this activity.

3. OP/0/A/1506/01 (Fuel & Component Handling) was being performed from memory because it was an Information Use procedure and was not required to be at the job location. Performing procedures from memory will increase the risk of human error. Requirements of OP/0/A/1506/01 were not met in that:

    a) The Control Room was not specifically notified that fuel handling was in progress in the Spent Fuel Pool (SFP).

    b) Fuel Handler A rotated the mast 90 degrees and back at the request of Reactor Engineer A. This was performed while the FA was not "full up" in the mast.

    c) Steps to lower a FA and disconnect from the fuel grapple are included in the procedure but the omission of those steps resulted in FA-? being left suspended inside the fuel mast.

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d) Limit and Precaution 2.22 directs that "When a Bridge is idle for 15 minutes or greater and the hoist is not engaged, turn off the Bridge hydraulic pump to prevent overheating." This condition was not met when Fuel Handler A secured the hydraulic pump because the hoist was engaged. Due to workarounds with the hydraulic pump and instruction from the Fuel Handling Supervisor, this had become a common fuel handling practice.

4. Inadequacy of OP/0/A/1506/01 (Fuel & Component Handling) in that it did not provide steps for the fuel handler to verify that the fuel bridge mast was empty prior to shutting down the bridge.

5. The failure to provide an adequate pre-job briefing for the evolution.

The pre-job briefing did not address roles and responsibilities of the individuals involved. During most of the activities, Fuel Handler A was acting under the direction of Reactor Engineer A. This potentially led to an expectation on the part of Fuel Handler A for Reactor Engineer A to instruct him to lower the FA. Reactor Engineer A felt it was not his responsibility to ensure that FA-7 was lowered back into the SFP racks.

Past industry and site experience was reviewed to determine if this event is recurring. It was concluded that industry operating experience has not been used effectively at Oconee to prevent fuel handling events. SER 91-15, as an example, identified fuel mispositioning events that occurred within the industry due in part to inadequate independent verification and self-verification techniques. Oconee reviewed the SER, revised refueling procedures, enhanced methods of fuel handlers communication, and evaluated training in response to this SER. However, these corrective actions were ineffective in preventing four fuel mispositioning events that occurred in 1992 through 1994.

An operating experience review was performed using the Oconee Problem Investigation Process (PIP) data base in the area of fuel handling activities to look for similar events with root causes similar to this event. Attachment A to this report summarizes past fuel handling events and the related NRC violations.

The first root cause (self-verification as it relates to fuel handling work practices) has contributed to four events resulting in three NRC violations at Oconee during the period of 1992 through 1995.

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The second root cause (lack of management expectations for formality in all aspects of the fuel handling process) has also contributed to fuel handling events at Oconee (particularly PIP 1-094-0707 and the associated NRC violation of August 2, 1994).

Therefore, it is concluded that this event is recurring with respect to both root causes. The repetitive nature of these fuel handling events demonstrate the lack of full use of lessons learned from previous events and application of too narrow a scope for corrective actions.

There were no radioactive releases, personnel injuries or over exposures, or NPRDS reportable equipment problems associated with this event.

CORRECTIVE ACTIONS

Immediate

1. Fuel Handlers lowered the fuel assembly into a Spent Fuel Pool (SFP) storage rack location.

2. Mechanical Maintenance management suspended fuel handling activities pending procedure changes.

Subsequent

1. Engineering calculations were performed and this event was analyzed with respect to the potential for exceeding design basis releases.

Planned

1. Step by step procedures will be required for all fuel movements.

2. A procedure checklist will be provided to assure that the fuel mast is returned to a proper end state at the conclusion of fuel handling.

3. Formalized pre-job briefings for all fuel related activities in the SFP will be implemented.

4. Appropriate personnel corrective actions will be taken in accordance with Duke Power policies.

5. A Self Initiated Technical Audit (SITA) will be performed to provide a broader review of fuel handling and other SFP activities and work processes.

Planned corrective actions 1 through 5 are considered Commitments to the NRC. They are the only items included in this report intended to be NRC Commitments.

SAFETY ANALYSIS

The consequences of the failure of a fuel assembly (FA) in the spent fuel pool (SFP) are analyzed in the Final Safety Analysis Report (FSAR), Section 15.11.2.1, "Single Fuel Assembly Handling Accidents". The FSAR accident scenario is a radioactive release from all 208 fuel rods. This accident is assumed to occur under at least 9 feet of water for iodine retention. The dose calculation with the FSAR initial condition assumptions of release inventories and conditions yields a dose of .66 rem whole body and 174 rem thyroid at the site boundary.

During an event requiring the Standby Shutdown Facility (SSF) Reactor Coolant (RC) makeup pump, FA NJ06E7 (FA-7) would have been uncovered by the decreasing inventory of the SFP. A heat up calculation of air cooling of the FA has been performed using the actual decay time after shutdown assuming only radial free convection and radiation. Results indicate a maximum cladding wall temperature at the top of the FA of 1022 degrees F. Potential damage mechanisms and the applicable limiting temperatures are:

cladding creep out (ballooning) and rupture	1150 deg F.
accelerated oxidation	1600 deg F.
metal water reaction	2200 deg F.
enhanced fission gas release from the UO2 pellet matrix	2450 deg F.
zircaloy melting	3400 deg F.

This calculation shows that cladding integrity would be maintained and no effluent radiation release occurs. Therefore, the existing analysis in Section 15.11.2.1 is still bounding.

An estimation was also performed for the most limiting decay heat load possible. In this case a high powered assembly, only 72 hours after subcriticality, was assumed in the mast and cooled by air and radiation. This analysis determined a maximum cladding temperature of 2000 degrees F. In this scenario, damage to the cladding would occur, and there would be no iodine retention in water, so the release of radiation from the assembly would be significant.

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Depletion of the SFP inventory removes the majority of the shielding from the spent fuel assemblies such that direct radiation shine from the spent fuel will become significant. However, the SFP walls provide lateral shielding so the direct radiation shine is primarily in a vertical direction. Since the top of FA-7 was approximately 9 feet below the SFP grade, this will only add a small amount of additional direct radiation to either the on-site or off-site dose rate.

Since the SFP inventory must be eventually replenished remotely, having FA-7 in the fuel mast does not impose any additional restrictions to the operability of the SSF RC makeup system.

During the time period of interest, no spent fuel was moved in the SFP. Since the fuel mast provides a positive mechanical lock for the spent FA and the SFP bridge is seismically designed, no additional potential for a fuel handling accident existed.

Using the updated Oconee PRA model, the annual frequency of an event relying on the standby shutdown facility for core damage mitigation is 3.3 E-04. For the 25 day period FA-7 was in the fuel mast, the probability becomes 2.3 E-05. Furthermore, typical PRA calculations utilize a 24 hour minimum time for the system relied upon to mitigate the accident. In this case a time in the range of 36-40 hours would have been available before the SFP inventory is depleted to a level exposing a portion of the FA.

In conclusion, during the period from Dec 14, 1995 to Jan. 8, 1996, when FA-7 was suspended in the fuel mast, FA-7 was in a static, stable position such that the probability of fuel damage by another mechanism (collision, dropped object, seismic event, etc.) was remote. No SSF event occurred during this period. FA-7 was not damaged and did not release any radioactive materials to the public. In the unlikely happenstance that a SSF event actually did occur, an extensive period of up to 36 to 40 hours would have been available for compensatory actions to be taken prior to uncovering FA-7. Additionally, calculations show that FA-7 would have been adequately air cooled and no damage would be expected. Therefore, the health and safety of the public was not affected by this event.

ATTACHMENT A

OPERATING EXPERIENCE REVIEW

Oconee Fuel Placement Events

PIP# *	Description
1-092-0723	Wrong fuel assembly (FA) was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Changes to the refueling procedure were implemented as corrective actions to prevent recurrence.
1-092-0724	Wrong FA was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Changes to the refueling procedure were implemented as corrective actions to prevent recurrence
1-094-0707	Refueling sequence was altered at the request of reactor engineers to observe nuclear instrumentation response without proper documentation and procedural control. This was a non-conservative decision made by the SRO in charge of fuel handling, Reactor Engineer, and the Fuel Handling Supervisor. Corrective actions to prevent recurrence involved a change in the refueling procedure to prohibit sequence deviations without the use of a procedure change or test procedure.
1-094-0714	A wrong FA was placed into Unit 1 Reactor Core during refueling activities as a result of inadequate self-check and independent verification. Corrective actions to prevent recurrence involved changes to procedures and methods of independent verification.

* PIP = PROBLEM INVESTIGATION PROCESS

Oconee Fuel Movement Events/Concerns

PIP#	Description
2-092-0024	A FA and control rod was damaged while the FA was being positioned for repair. The procedure was not reviewed prior to the move and the control rod and FA were damaged. Corrective actions to prevent recurrence involved procedure changes and pedestal modifications.
3-092-0470	Bent spider assemblies causes delay in removal of burnable poison rods from two fuel assemblies. It could not be determined whether the damage occurred as a result of previous fuel handling activities by Duke or by the fuel vendor. Corrective actions involved manufacturing a component sizing template to be used by Quality Assurance during the component inspection performed upon unloading of the new fuel assemblies.
2-093-0431	An intermediate grid strap became torn and separated from its FA during refueling operation. This type of damage is caused when the grid straps of adjacent assemblies snag each other during fuel movements made in the core. Corrective actions to prevent recurrence involved changes to the refueling procedure to provide new guidance to prevent FA grid strap damage.
3-094-0204	A dummy control rod assembly located in the deep end of the fuel transfer canal was struck while transporting the core support assembly. This was a result of inadequate self-check of clearances. Crane control and water clarity problems contributed to the problem. Transport had to be halted to perform inspections of the core support assembly, the transfer canal liner plate, and the fuel storage racks. Corrective actions to prevent recurrence involved procedure changes to incorporate preventive measures.
1-095-1429	During reactor defueling activities, Spent Fuel Pool (SFP) bridge hoist and grapple operation was hampered several times due to unexpected interference with consolidated fuel canisters. This interference problems in disengaging from fuel assemblies. Corrective actions involved moving the consolidated fuel canisters to an area of the SFP that is outside of the off-load area.
[illegible]-095-1462	FA NJ0776 was found to have significant structural damage on four consecutive intermediate spacer grids on the southwest corner. No fuel rod damage was found or suspected.

INDUSTRY OPERATING EXPERIENCE

PIP#	Description
1-C95-0184	During set up of the B&W Fuel Reconstitution, elevator parts sheared/fell from the elevator into the cask area. The elevator part apparently sheared when it contacted the cask area wall. Elevator design deficiency and worker attention to detail contributed to this event.
1-M93-0055	Several new fuel assemblies were received and placed in storage cells that were not in accordance with procedure. Root causes were failure to follow procedure and inattention to detail. The fuel components were not adversely affected.
1-M93-0414	A control rod was inserted in the wrong FA. Corrective actions involved procedure changes and personnel training to prevent recurrence.
2-M93-0676	A contractor personnel failed to follow procedural requirements for handling fuel rods during reconstitution activities, which resulted in severely bent fuel rod and subsequent challenge to the fuel cladding integrity. This resulted in a NRC level IV violation (PIP 2-M93-0917) for the failure of contractor personnel to follow procedural requirements.
1-M96-0002	A sequence within the FA-Insert shuffle procedure was performed incorrectly resulting in the misposition of a thimble plug in the SFP. The verification process identified and corrected this discrepancy. No corrective actions to prevent recurrence were identified.
SER 91-15	This report describes six industry fuel mispositioning events during refueling and defueling activities as a result of inadequacies in procedures, independent verification, and training. Oconee's review of this event resulted in changes to refueling procedure changes and methods of communication
SER 94-4	This report describes six specific industry events that involve human performance deficiencies while handling reactor core components that resulted in actual FA or other core component damage, damage to refueling equipment, and/or increased potential for damage to fuel or other core components

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit One	05000 269	96	- 01	- 00	16 OF 17

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Oconee incorporated these industry events and lessons learned into the operations fuel handling lesson plan.

IN 94-13 This report describes potential problems resulting from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel based on four industry events.

Oconee's review of this report resulted in no recommended actions based on actions taken with SER 94-4.

IN 94-13, Sup. 1 This report describes an industry event involving unauthorized movement of a defective spent fuel rod.

Oconee's review of this report resulted in no recommended actions.

OCONEE NRC INSPECTIONS

NRC Level IV Violation (November 21, 1990)

One example of a failure to adequately implement a refueling procedure that resulted in a FA being placed in the wrong location in the core. Root causes were operator error and poor visibility in the SFP. Corrective actions to prevent recurrence involved counseling the bridge operator.

NRC Level IV Violation (September 17, 1991)

One example of failure to adequately implement a refueling procedure that resulted in a FA being placed in the wrong spent fuel location. Root causes were insufficient attention to detail, insufficient procedure detail and communication errors. Corrective actions to prevent recurrence involved procedural changes and fuel handling training

NRC Level IV Violation (February 22, 1991)

NRC FORM 366A

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

TEXT CONTINUATION

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Oconee Nuclear Station, Unit One	05000 269	96	-- 01	-- 00	17 OF 17

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Two examples of failure to implement refueling procedures that resulted in two fuel assemblies being placed in the wrong location in the core. Root cause was inadequate self-checking. Corrective actions to prevent recurrence involved procedural changes. (Covered by PIPs 1-092-0723 and 1-092-0724)

NRC Level IV Violation (August 2, 1994)

Refueling sequence was altered to observe nuclear instrumentation response without proper documentation and procedural control. This was performed at the request of Reactor Engineering personnel. (Covered by PIP 1-094-0707)

NRC Level IV Violation with Civil Penalty (August 2, 1994)

A FA retrieved from the wrong spent fuel location and placed in the reactor core. Root causes were inadequate self-check and independent verification. This was the fourth occurrence of failure to identify and adequately verify FA locations. Corrective actions to prevent recurrence involved procedural changes and personnel training. (Covered by PIP 1-094-0714 and PIP 1-094-0707)

EXHIBIT B-15

Oyster Creek Unit 1:
LER 219/87-006-00 (February 24, 1987)

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Oyster Creek, Unit 1	0 5 0 0 0 2 1 9	1 OF 0 3

TITLE (4) TECHNICAL SPECIFICATION VIOLATION CAUSED BY IMPROPER STORAGE OF HIGHER ENRICHMENT FUEL DUE TO PERSONNEL ERRO

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 1	2 1	8 7	8 7	0 0 6	0 0	0 2	2 4	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 2 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

[ ] 20.402(b)	[ ] 20.405(c)	[ ] 50.73(a)(2)(iv)	[ ] 73.71(b)
[ ] 20.405(a)(1)(i)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(v)	[ ] 73.71(c)
[ ] 20.405(a)(1)(ii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(vii)	[ ] OTHER (Specify in Abstract below and in Text, NRC Form 366A)
[ ] 20.405(a)(1)(iii)	[X] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(A)	
[ ] 20.405(a)(1)(iv)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(viii)(B)	
[ ] 20.405(a)(1)(v)	[ ] 50.73(a)(2)(iii)	[ ] 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Hari S. Sharma, Core Engineer	6 0 9 9 7 1 - 4 6 3 8

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
[X] YES (If yes, complete EXPECTED SUBMISSION DATE)	[ ] NO		1 2	3 1	8 7

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

Oyster Creek Technical Specification 5.3.1(C) specifies that the fuel stored in the fuel pool storage racks shall not exceed a maximum average planar enrichment of 3.01 wt% U-235. Contrary to the above, reload fuel bundles supplied by General Electric Company (GE) having an average planar enrichment of 3.19% U-235 were temporarily stored in the fuel pool during the 11R outage in 1986. The cause of the event is attributed to personnel error in not performing a thorough safety analysis for storage of the new fuel and in not recognizing a conflict with the Technical Specifications prior to fuel storage in the spent fuel pool.

Corrective actions will consist of revising the refueling procedures, revising the Technical Specifications to raise the enrichment limitations on stored fuel, and reviewing the occurrence with engineering personnel.

DATE OF DISCOVERY

The violation was discovered on January 21, 1987 during a subsequent review of the Oyster Creek Technical Specifications for potential changes related to the new fuel design.

IDENTIFICATION OF OCCURRENCE

Fuel with an average planar enrichment of 3.19 wt% U-235 was stored in the spent fuel pool beginning February 27, 1986. Technical Specification 5.3.1(C) states that the fuel to be stored in the spent fuel storage facility shall not exceed maximum average planar enrichment of 3.01 wt% U-235. This event is reportable under 10CFR50.73 (a)(2)(i)B.

CONDITIONS PRIOR TO DISCOVERY

At the time of occurrence, the plant was operating in a coastdown mode in preparation for the 11R outage. At the time of discovery, the plant was at approximately 20% power starting up for Cycle 11 operation. All the fuel bundles which exceeded the Technical Specification enrichment limitations for storage in the fuel pool had been removed from the spent fuel pool and loaded in the core.

DESCRIPTION OF OCCURRENCE

A total of 204 GE P8DRB299 fuel bundles, with an average planar enrichment of 3.19 wt% U-235 and a bundle average enrichment of 2.99 wt% U-235, were received in 1986. At the time of fuel receipt, the dry storage vault had a capacity for 140 bundles. Initially, 64 of the new bundles were temporarily stored in the spent fuel pool. As the outage progressed, more bundles were taken out of the dry storage vault, channelled and stored in the spent fuel racks. Ultimately, 184 reload assemblies were subsequently stored in the spent fuel pool prior to the start of core reload in August 1986. At the end of core reload (September 14, 1986), all the P8DRB299 fuel in the spent fuel pool had been transferred to the core.

APPARENT CAUSE OF OCCURRENCE

The cause of this occurrence is attributed to personnel error. The safety analysis which was prepared was orienten toward the safe operation of the plant using the higher enrichment fuel during the next cycle. It did not take into account that the new fuel could conceivably be stored in the spent fuel pool (only dry storage was considered). Had this possibility been envisioned, the need for a Technical Specification change would have been recognized.

A contributing factor in this event is that procedural controls were inadequate. The refueling procedure (205.0) contains a precaution regarding the Technical Specification restriction on fuel bundle enrichment, however, the refueling procedures do not require verifications to ensure compliance with the enrichment restriction associated with fuel stored in the spent fuel pool. Had such a verification been performed, the fuel would not have been stored in the spent fuel pool.

ANALYSIS OF OCCURRENCE AND SAFETY ASSESSMENT

The initial criticality analysis of the High Density Poison Racks (HDPR) assumed a uniformly enriched lattice of 3.01 wt% U-235. The analysis also did not take credit for burnable poisons as allowed in Regulatory Guide 1.13. The lattice K-infinity for the analysis was determined to be 1.33 which resulted in a HDPR cell K-effective of less than 0.95. The P8DRB299 bundles in the spent fuel pool have a maximum cold, uncontrolled K-infinity of 1.22 as determined by the fuel vendor. Therefore, it is expected that the cell K-effective did not reach or exceed 0.95. However, a re-evaluation of the HDPR criticality analysis is currently being performed taking credit for burnable poisons. The results of this analysis will be submitted in a supplement report.

Corrective Actions

Currently, there are no fuel bundles with an average planar enrichment of greater than 3.01 wt% U-235 in the spent fuel pool. Corrective actions will consist of the following:

1. Fuel movement procedures will have appropriate controls added that ensure Technical Specification compliance in this area.

2. Based upon the results of the HDPR re-evaluation, a Technical Specification change request will be submitted to allow fuel bundles with higher average planar enrichments to be stored in the spent fuel pool.

3. This event will be reviewed with the engineering personnel involved stressing the requirements to consider all licensing basis documents and associated restrictions when performing safety reviews.

SIMILAR OCCURRENCES

None

(0288A)

Nuclear

GPU Nuclear Corporation
Post Office Box 388
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Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number:

February 24, 1987

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Licensee Event Report

This letter forwards one (1) copy of Licensee Event Report (LER) No. 87-006.

Very truly yours,

Peter B. Fiedler
Vice President and Director
Oyster Creek

PBF:KB:dam(0288A)
Enclosures

cc: Dr. Thomas E. Murley, Administrator
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U.S. Nuclear Regulatory Commission
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NRC Resident Inspector
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Nuclear

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EXHIBIT B-16

NRC Information Notice 94-13
(February 22, 1994)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 22, 1994

NRC INFORMATION NOTICE 94-13: UNANTICIPATED AND UNINTENDED MOVEMENT OF FUEL ASSEMBLIES AND OTHER COMPONENTS DUE TO IMPROPER OPERATION OF REFUELING EQUIPMENT

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potential problems resulting from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Vermont Yankee Events

The Vermont Yankee facility was in a refueling outage with fuel movement in progress on September 3, 1993, when an irradiated fuel assembly became detached from the grapple after being lifted out of its position in the reactor core. The assembly fell approximately 2.4 m [8 ft] back into its original location in the reactor core. The licensee suspended fuel handling and investigated the event. The licensee determined that the grapple had not properly engaged the lifting bail on the fuel assembly and that the personnel performing the fuel handling activities had failed to verify proper grapple engagement. After completing the investigation and taking corrective actions, the licensee resumed fuel handling activities on September 7, 1993.

On September 9, 1993, a fuel assembly that was being moved to a fuel sipping can was inadvertently lowered, instead of raised, striking another core component. The potentially damaged fuel assembly was then moved to the fuel sipping can and the licensee again suspended fuel handling activities. The NRC dispatched an augmented inspection team (AIT) on September 9, 1993, to investigate the fuel handling incidents.

The AIT documented its findings in NRC Inspection Report 50-271/93-81, issued October 21, 1993. The AIT concluded that mistakes made by refueling personnel were the immediate causes of both events. In addition, weaknesses in the human factors aspects of the controls for the fuel handling equipment [illegible] for the event in which a fuel assembly was lowered rather than [illegible] controls for the fuel handling equipment had been modified [illegible] this event occurred. The [illegible] concluded that the root cause of [illegible] was a significant weakness in management oversight of fuel handling activities. [illegible] management oversight had allowed many of the measures [illegible] to prevent fuel handling accidents to be neglected. The AIT found that (1) design changes were not transmitted to all in a timely and accurate [illegible] modifications to the refueling bridge, (2) procedures were not always used and, when they were used, they were not always adhered to, and (3) supervisors did not ensure that procedures were followed. In addition, the AIT found that training was not effective in that operators were not aware of certain key procedure steps in most instances. Specifically, the personnel monitoring the fuel handling activities were not aware of the requirement to visually verify grapple closure when engaging and lifting fuel assemblies. The AIT found that management did not communicate expectations and provide proper oversight of fuel handling activities.

Peach Bottom Events

With Unit 3 shut down for refueling on September 23, 1993, a fuel assembly could not be fully inserted into its spent fuel rack cell. It was thought that the fuel assembly had swelled due to irradiation in the core, and the fuel assembly was successfully placed in a different cell. It was further postulated that there might be some debris in the cell, and that the cell should be checked at some future date. On September 24, 1993, another fuel assembly became stuck in its spent fuel rack cell. The licensee evaluated the material condition of the fuel assembly, calculated an allowable lifting force, and conferred with the fuel vendor. The licensee increased the load limit of the refueling hoist and the fuel assembly was freed from the rack with no damage to the fuel assembly. Subsequent examinations revealed that sections of local power range monitor instrument strings that had previously been cut up were in the bottoms of three cells in the rack, including the two cells with which difficulties were experienced. The licensee believes that the debris may have fallen into the cells during a fuel pool cleanup effort conducted during the previous summer.

The licensee is currently investigating why the debris was in the spent fuel pool and why the refueling personnel did not ensure that the spent fuel rack cells did not contain any debris prior to inserting the fuel assemblies.

Susquehanna Events

The Susquehanna Steam Electric Station Unit 1 was shut down with defueling in progress on October 6, 1993, when the personnel performing the fuel handling activities removed an incorrect fuel assembly from a peripheral location in the core. The personnel involved realized they had removed the wrong assembly and they inappropriately decided to return the assembly to its prior position in the core. The appropriate action, per licensee procedures, would have been to place the bundle in the spent fuel pool and secure fuel handling activities until the cause of the error was determined and corrected.

On October 26, 1993, while lowering a fuel assembly into the core during refueling, an unexpected drop of 25 to 38 cm [10 to 15 in] of one of the sections of the fuel handling mast occurred. The fuel assembly was not damaged and it did not strike the vessel internals. Subsequent testing [illegible]ced mechanical binding of the mast and a bend in the mast was [illegible]d. The binding [illegible] one section of the mast while a [illegible] extended. Eventually [illegible] section [illegible] limited distance. The licensee subsequently determined that the mast had been bent by impact with the flange protector on the reactor vessel while traversing through the "cattle chute" between the [illegible]d fuel pool because the mast was not raised high enough. The Unit 2 refueling bridge was transferred to Unit 1 and, after satisfactory completion of surveillance testing, refueling was resumed.

On October 27, 1993, while transferring a double blade guide to the spent fuel pool, the blade guide hit the side of the reactor vessel because it was not raised high enough to clear the vessel. The licensee suspended refueling activities, revised the associated procedure, and inspected the mast. The core reload was resumed after surveillances on the fuel handling equipment were successfully conducted. On October 28, 1993, while attempting to grapple a new fuel assembly in the fuel pool, the personnel performing the fuel handling activities heard two loud bangs and observed bubbles in the pool for 5 to 10 seconds. Subsequent inspection revealed that one section of the mast from Unit 2 was bent. The licensee believes that the mast was weakened by the impact with the reactor vessel that occurred during the October 27 event.

On October 29, 1993, the NRC dispatched an AIT to the site to review the events. The AIT documented its findings in Inspection Report 50-387/93-80, issued on December 21, 1993. The AIT concluded that facility management did not maintain proper oversight of refuel floor activities and that inadequate corrective actions were implemented in the past for problems with the fuel handling equipment. The AIT also concluded that the licensee fuel handling procedures were adequate for the proper completion of the fuel handling activities, although certain improvements could be made to increase the awareness of the operators concerning potential problems.

<u>Nine Mile Point Event</u>

Nine Mile Point Unit 2 was shut down with refueling in progress on November 1, 1993, when a blade guide was moved from the core into the spent fuel pool. The contractor refueling operator disengaged the grapple and observed the correct light indication on the bridge. There was no procedural requirement to visually verify disengagement or for the Senior Reactor Operator Limited to Fuel Handling (LSRO) or the spotter to verify disengagement. The refueling operator noticed increased drag after the refueling bridge crane had been moved approximately 23 cm [9 in] toward the next location. At that time, licensee personnel determined that the blade guide was still engaged on the grapple. The bridge was returned to its previous position, the blade guide was lowered and disengaged (positive verification was obtained this time), and the operator proceeded to move the next component, which was a fuel assembly. While lowering that fuel assembly [illegible] the refueling operator noticed that the mast was binding. At [illegible] became involved and directed that the fuel assembly be [illegible] pool. While lowering the fuel assembly into the rack in [illegible] of the mast dropped [illegible] and 79 [illegible] the fuel assembly [illegible] disengaged [illegible] subsequently determined that the mast [illegible] was damaged. After the licensee [illegible] and repaired the fuel handling equipment, fuel [illegible] on November [illegible], 1993.

[illegible] determined that there were several personnel performance issues [illegible] addressed. The refueling operator had been trained to verify disengagement after releasing each component, although the procedure only required verification of ungrappling when handling fuel assemblies. Disengagement was to be verified by raising and rotating the mast. The refueling operator did not verify disengagement after releasing the blade guide. In addition, the refueling operator did not notify the LSRO of the unanticipated equipment response (remaining connected to the blade guide while traversing the bridge). Also contributing to the event was the fact that the LSRO was observing a refueling bridge trolley bearing about which he was concerned, rather than the handling of the blade guide. Licensee review determined that management expectations regarding the supervision of refueling activities had not been clearly expressed to the LSROs.

Discussion

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. Either of these situations could require increased management attention and oversight by the licensee to ensure proper and safe performance of fuel handling activities.

Appendix B to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50) requires licensees to have appropriate procedures to control activities affecting quality (such as the actions to be taken during operation of refueling equipment), and that the procedures are used and followed. In addition, 10 CFR 50.120 requires licensees to implement a training program for various categories of nuclear power plant personnel to ensure that those personnel have the necessary knowledge, skills, and abilities to perform their assigned jobs competently. This rule applies to the personnel (including contractors) who operate or supervise the operation of the refueling equipment. The cases discussed in this notice include situations in which the licensees failed to conduct appropriate training in the use of their refueling equipment, particularly with respect to design modifications made to the controls for the fuel mast. These events also demonstrate that the fuel handling personnel involved in certain instances were variously not aware that management expected them to identify deviations from expected results, cease operations when an unexpected or abnormal condition is encountered, and notify operations and/or plant management of unexpected or abnormal conditions.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below, or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contacts: P. L. Eng, NRR
(301) 504-1837

E. M. Kelly, RI
(215) 337-5183

J. R. White, RI
(215) 337-5114

L. E. Nicholson, RI
(215) 337-5128

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
94-12	Insights Gained from Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment	02/09/94	All holders of OLs or CPs for nuclear power reactors.
94-11	Turbine Overspeed and Reactor Cooldown during Shutdown Evolution	02/08/94	All holders of OLs or CPs for nuclear power reactors.
94-10	Failure of Motor-Operated Valve Electric Power Train due to Sheared or Dislodged Motor Pinion Gear Key	02/04/94	All holders of OLs or CPs for nuclear power reactors.
94-09	Release of Patients with Residual Radioactivity from Medical Treatment and Control of Areas due to Presence of Patients Containing Radioactivity Following Implementation of Revised 10 CFR Part 20	02/03/94	All U.S. Nuclear Regulatory Commission medical licensees.
94-08	Potential for Surveillance Testing to Fail to Detect an Inoperable Main Steam Isolation Valve	01/01/94	All holders of OLs or CPs for nuclear power reactors.
93-26, Supp. 1	Grease Solidification Causes Molded-Case Circuit Breaker Failure to Close	01/31/94	All holders of OLs or CPs for nuclear power reactors.
94-07	Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under the Revised 10 CFR Part 20	01/28/94	All byproduct material and fuel cycle licensees with the exception of licensees authorized solely for sealed sources.

OL = Operating License
CP = Construction Permit

EXHIBIT B-18

NRC Information Notice 94-13, Supplement 1
(June 28, 1994)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

June 28, 1994

NRC INFORMATION NOTICE 94-13, SUPPLEMENT 1: UNANTICIPATED AND UNINTENDED MOVEMENT OF FUEL ASSEMBLIES AND OTHER COMPONENTS DUE TO IMPROPER OPERATION OF REFUELING EQUIPMENT

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice supplement to alert addressees to an event involving unauthorized movement of a defective spent fuel rod. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

The NRC issued Information Notice (IN) 94-13, "Unanticipated and Unintended Movement of Fuel Assemblies and Other Components Due to Improper Operation of Refueling Equipment," to alert addressees to problems that could result from inadequate oversight of refueling operations and inadequate performance on the part of refueling personnel. IN 94-13 described various refueling events that occurred at Vermont Yankee, Peach Bottom, Susquehanna, and Nine Mile Point. These events demonstrate the importance of proper controls over, and operation of, refueling equipment during use. A recent event at the Waterford Steam Electric Station (Waterford) demonstrates the potential for fuel damage or personnel hazards which could result from fuel-handling equipment that is not properly stored and not secured from unauthorized use.

Description of Circumstances

On February 18, 1994, the Waterford plant was operating at 100-percent power when a senior reactor operator found an unknown object hanging from the fuel-handling machine in the fuel-handling building. Health physics technicians measured radiation levels in the spent fuel pool area and found them to be normal. Licensee personnel remotely secured the object with vise grips and determined that underwater radiation levels were .2 to .7 Sv/hr [20 to 70 R/hr] at 15 centimeters [6 inches] from the object. A Combustion Engineering employee identified the object as a fuel rod encapsulation tube. No visual damage was apparent on the tube. The licensee posted a security guard in the spent fuel pool area and reported the event to the NRC.

The licensee reviewed fuel storage records and determined that the tube contained a defective fuel rod that had been removed from an irradiated fuel assembly several years earlier. At that time, the tube had been placed in a center guide tube in a grid cage stored in the spent fuel racks. The licensee reviewed computer access records for the fuel-handling area and interviewed relevant personnel about the event. Personnel who may have had access to the fuel-handling machine completed questionnaires regarding the event. The licensee determined that the refueling director had used the fuel-handling machine the day before the object was discovered and had parked the fuel-handling machine at a location directly over the fuel rod encapsulation tube. However, the refueling director had not used the hoist and was not sure that he would have noticed if the encapsulation tube was hanging from the hoist at the time he used the machine. Surveillance records indicated that the fuel rod encapsulation tube must have become attached to the fuel-handling tool sometime between February 11 and 18, 1994.

Design drawings of the cap of the fuel rod encapsulation tube showed that the outer diameter of the cap was about equal to the inner diameter of the end of the fuel-handling tool. Apparently, the cap had become bound in the fuel-handling tool when the hoist was lowered to the top of the spent fuel rack and, when the hoist was raised, the tube was completely removed from the grid cage.

Although contractors had performed the fuel-handling operations for previous refueling outages, Waterford personnel were scheduled to perform the fuel handling for the March 1994 refueling outage. The licensee speculated that one of the people assigned to fuel-handling activities for the March outage may have inadvertently lifted the encapsulation tube while practicing the use of the hoist. Personnel were required to notify health physics staff before accessing the refueling machine; however, health physics records showed that no one had made such a notification during this time. No keys or special knowledge was needed to access the controls of the fuel-handling machine. Electrical power could be obtained by closing two electrical breakers and pushing one switch that were located on the machine. The licensee questioned several employees, but no one admitted to unauthorized use of the fuel-handling machine.

As an interim corrective action, the licensee deenergized the computer that controls the fuel-handling machine by opening a breaker in a locked power control center. The licensee planned to (1) develop a means to prevent the fuel rod encapsulation tube from being inadvertently lifted by the fuel-handling tool, (2) add a precaution to the operating procedure warning operators not to lower the fuel-handling tool over the storage location, and (3) add hoist manipulations to the lesson plans for proficiency training.

Discussion

Procedures governing the use of equipment for handling fuel and core components may not prevent unauthorized or unintended operation of that equipment. Precautions such as locking out breakers that energize the fuel-handling equipment and the placement of placards in highly visible areas declaring that unauthorized operation of fuel-handling equipment is forbidden may help ensure that the equipment is not used without proper authorization. Additionally, storing the fuel-handling machine in an area where accidental movement of the hoist or grapple will not impact stored fuel or other components may contribute to the prevention of inadvertent fuel movement or damage. Management attention and oversight of the operation of fuel and core component handling equipment is important to ensure that fuel and core components are protected from damage or unauthorized movement and that plant personnel are protected from unnecessary exposure to radiation.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Alfred E. Chaffee for

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: Dale A. Powers, RIV
(817) 860-8195

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
94-47	Accuracy of Information Provided to NRC during the Licensing Process	06/21/94	All U.S. Nuclear Regulatory Commission Material Licensees.
94-46	NonConservative Reactor Coolant System Leakage Calculation	06/20/94	All holders of OLs or CPs for nuclear power reactors.
94-45	Potential Common-Mode Failure Mechanism for Large Vertical Pumps	06/17/94	All holders of OLs or CPs for nuclear power reactors.
94-44	Main Steam Isolation Valve Failure to Close on Demand because of Inadequate Maintenance and Testing	06/16/94	All holders of OLs or CPs for nuclear power reactors.
94-43	Determination of Primary-to-Secondary Steam Generator Leak Rate	06/10/94	All holders of OLs or CPs for pressurized water reactors.
94-42	Cracking in the Lower Region of the Core Shroud in Boiling-Water Reactors	06/07/94	All holders of OLs or CPs for boiling-water reactors (BWRs).
94-41	Problems with General Electric Type CR124 Overload Relay Ambient Compensation	06/07/94	All holders of OLs or CPs for nuclear power reactors.
94-40	Failure of a Rod Control Cluster Assembly to Fully Insert Following a Reactor Trip at Braidwood Unit 2	05/26/94	All holders of OLs or CPs for pressurized-water reactors (PWRs).
94-39	Identified Problems in Gamma Stereotactic Radiosurgery	05/31/94	All U.S. Nuclear Regulatory Commission Teletherapy Medical Licensees.

OL = Operating License
CP = Construction Permit

EXHIBIT B-17

Three Mile Island Unit 1:
LER 289/98-002-01 (April 3, 1998)

GPU NUCLEAR

GPU Nuclear, Inc.
Route 441 South
Post Office Box 480
Middletown, PA 17057-0480
Tel 717-944-7621

April 03, 1998
~~1920-98-20150~~

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

S

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Licensee Event Report (LER) No. 98-002, Revision 1

On February 4, 1998, GPU Nuclear determined that the Spent Fuel Pool was not sampled in accordance with the requirements of the Technical Specifications Surveillance Requirement (SR) specified in Table 4.1-3, item 4, which requires sampling monthly and after each makeup. A review of work activities determined that no sample was taken following a water addition on January 23, 1998. This condition was found to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications. A subsequent analysis determined that the filling activity could not have diluted the boron concentration significantly.

This condition was reported to the NRC by letter dated March 3, 1998. Attached is Revision 1 of LER 98-002, which provides additional information that addresses the following items: the reason for this event, the extent of the problem associated with the missing operator aid, the assessment of the safety consequences and implications of the event, and the corrective action section.

The event did not affect the health and safety of the public.

Please contact Adam Miller, TMI Licensing at (717) 948-8128 if you have any questions regarding this matter.

Sincerely,

for

James W. Langenbach
Vice President and Director, TMI

I/

IE22

AWM

cc: TMI Senior Resident Inspector
Administrator, Region I
TMI-1 Senior Project Manager
File 98048

appropriate because no major replenishment of pool water is expected to take place over a short period of time." This bases appears to be consistent with the TMI-1 TS bases. The TMI-1 staff is evaluating if a request to revise the current surveillance requirement is appropriate.

VII. Corrective Actions:

A. Corrective Actions Taken:

1. A new Operator Aid has been posted at the valve that is used to fill the Spent Fuel Pool (SFP) from the Reclaimed Water System. **In order to ensure the Shift Supervisor tracks the need for the water sample, the new Operator Aid has been modified to add a step to require the individual doing the fill to notify the Shift Supervisor to track this item on the S/S Turnover until the SFP sample is taken and analyzed within the designated time period.**

2. The Primary Auxiliary Operator Turnover Checklist has been revised to include a requirement to notify the Chemistry Department of sample requirements if a water addition to the Spent Fuel Pool has either been initiated or completed during the shift. **The contents of this checklist are discussed at the crew briefing and the checklists of all the Operators are compiled and reviewed by control room supervision.**

B. Action Planned to Prevent Recurrence:

1. This **revised** LER will be reviewed by all of the appropriate personnel in the Operations and Chemistry Departments. The review will be documented and the documentation maintained by the Operations Department Administrator. This action will be completed within 60 days of the issuance of this **revised** LER.

2. **To determine the extent of the problem associated with missing Operator Aids, a spot check of Operator Aids will be performed. Each Shift Supervisor will select five (5) of the Operating Procedures for which he is the owner. This selection will only include procedures that contain Operator Aids. All of the Operator Aids contained in these 25 Operating Procedures (i.e. 5 crews at 5 procedures per crew) will be physically verified to insure that they are properly posted, not broken, legible, and accurate. This verification will be completed and an assessment of the verification performed by the Lead Operations Engineer prior to April 30, 1998. If the assessment reveals that the Operator Aids are in poor condition, a 100% verification will be performed for the remaining Operator Aids.**

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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Three Mile Island, Unit 1	05000289	98	-- 002 --	1	6 OF 6

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

3. **All of the Operating Procedures which contain Operator Aids will be reviewed to insure that the Operator Aids do not contain direction or guidance which would be the sole source of information provided during a task performance to comply with Technical Specification requirements. This review will be completed prior to April 30, 1998.**

4. **All licensed personnel will be given training on an overview of the contents of Technical Specifications section 4 and specific training on the sampling requirements of Table 4.1-3. This training will be completed by 12/31/98.**

* The Energy Industry Identification System (EIIS), System Identification (SI) and Component Function Identification (CFI) Codes are included in brackets, [SI/CFI] where applicable, as required by 10 CFR 50.73 (b)(2)(ii)(F).

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Three Mile Island, Unit 1	05000289	1 OF 6

TITLE (4)

MISSED SPENT FUEL POOL SAMPLE FOLLOWING A WATER ADDITION DUE TO UNFAMILIARITY WITH SAMPLING REQUIREMENTS AND A MISSING OPERATOR AID

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
2	04	98	98	002	1	04	03	98		
									FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Adam Miller, TMI Licensing Engineer	(717) 948-8128

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 4, 1998, GPU Nuclear determined that the Spent Fuel Pool was not sampled in accordance with the requirements of the Technical Specifications Surveillance Requirement (SR) specified in Table 4.1-3, item 4, which requires sampling monthly and after each makeup. A review of work activities determined that no sample was taken following a water addition on January 23, 1998. This condition was found to be reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by Technical Specifications (TS). An analysis determined that the filling activity could not have diluted the boron concentration significantly.

Contributing factors for this event **were Control Room supervision unfamiliarity with the sampling requirements and** a missing sign by the fill valve, which serves as an **Operator aid**. The missing sign has been replaced and the Primary Auxiliary Operator (AO) Turnover Checklist has been revised to include a requirement to notify the Chemistry Department of sample requirements if a water addition to the Spent Fuel Pool has either been initiated or completed during the shift. **Additionally, licensed personnel will be given training on Technical Specification section 4 requirements.**

There were no adverse safety consequences from this event, and the event did not affect the health and safety of the public.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

I. Plant Operating Conditions before Event:

TMI-1 was operating at 100% steady state power prior to and during the event described in this LER.

II. Status of Structures, Components, or Systems that were Inoperable at the Start of the Event and that Contributed to the Event:

None.

III. Event Description:

The TMI-1 Technical specifications Table 4.1-3.4 requires that the Spent Fuel Pool Water [DA]* be sampled monthly and after each makeup. Operating Procedure 1104-6 "Spent Fuel Cooling System" requires notification of the Chemistry Department at the completion of a Spent Fuel Pool water addition that a sample must be taken between 24 to 48 hours after the addition was completed.

Contrary to these requirements, a water addition was made to the Spent Fuel Pool on 01/23/98 (From 0918 to 1705) without the required follow-up water sample. Another addition was made to the Spent Fuel Pool on 01/27/98 (From 1410 to 1817). The Spent Fuel Pool was then sampled on 01/28/98 at 0430 and again on 01/29/98 at 0830. These samples exceeded the 48-hour sample requirement for the addition that was performed on 01/23/98.

During a routine review of work activities by the Chemistry Department, a Staff Chemist noticed that samples of the Spent Fuel Pool had been obtained on 01/28/98 and 01/29/98. He recognized that these samples were taken to comply with the requirement to sample the Spent Fuel Pool after each addition. The Staff Chemist identified a possible lack of formal tracking for samples after filling the pool to the Manager, Radwaste and Chemistry. The manager investigated the scope of the potential problem by reviewing two months of spent fuel pool boron concentration data and the computerized Control Room Logs. He found that there was no spent fuel pool boron data following an addition to the pool on 01/23/98. The manager submitted a CAP (Corrective Action Process) Form (T1998-0066) to document the missed sample.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

IV. Identification of Root Cause

The Primary AO was notified at the shift turnover meeting of the intent to fill the Spent Fuel Pool. This task is coordinated with processing (i.e. purifying) water with the ECOLOCHEM system on the secondary plant. As water is processed on the secondary plant it is transferred to the Reclaimed Water Storage Tank on the primary plant. This tank is then used as the source tank to fill the Spent Fuel Pool. Towards the end of the shift the AO was notified of the intent to shutdown the ECOLOCHEM system which would in turn require securing the filling of the Reclaimed Water Storage Tank and the Spent Fuel Pool. When the Primary AO terminated the filling of the Spent Fuel Pool he made an entry in his logbook and also notified the Control Room. He did not convey any sample requirement information to the Control Room. Contrary to the requirements of the Operating Procedure the Operations Department did not notify the Chemistry Department of the need to sample the Spent Fuel Pool and the Shift Supervisor did not track the need for the water sample on the Shift Supervisor's Turnover. **The Shift Supervisor was aware that the Spent Fuel Pool fill had been performed, but he was unaware of the Technical Specification section 4 sampling requirements.**

The task of filling the Spent Fuel Pool is considered a routine evolution that does not require the operator to actually use a copy of the Operating Procedure to perform the evolution. For this reason an Operator Aid is affixed to the wall directly behind the valve used to fill the Spent Fuel Pool in order to remind the Operator of the notification and sampling requirements. However, the Operator Aid was missing from the wall on 01/23/98 when the Spent Fuel Pool was filled. Therefore there was no Operator Aid available to serve as a reminder to the Operator to notify Chemistry and the Shift Supervisor at the completion of the fill process.

There has been one previous occurrence, June 13, 1996, where a water addition was made to the Spent Fuel Pool without the required follow-up sample being performed. (This is the first that resulted in an LER due to the recent change at TMI-1 concerning the reportability of a missed Tech Spec Surveillance). As a result of the previous occurrence, the procedure guidance contained in Operating Procedure 1104-6 "Spent Fuel Cooling System" was enhanced to require notification of the Chemistry Department of the required sample and to track the need for a sample on the Shift Supervisor's Turnover until the sample is taken and analyzed. As part of the procedure enhancement, an enclosure to the procedure was added to indicate that an Operator Aid is posted at the Reclaimed Water supply valve.

Factors which **contributed** to the failure to obtain the Tech Spec required sample of the Spent Fuel Pool following a water addition are:

- Pertinent information not transmitted
- Required procedure/document not followed
- **Installed Operator Aid not provided (i.e. missing)**

As stated these are all contributing factors. We incorporate a work ethic that relies on defense in depth to guard against errors. The last barrier in this defense is ideally the Control Room licensed supervision. In this case we inappropriately relied on an Auxiliary Operator with the use of an Operator Aid to provide the last barrier. The Control Room licensed supervision missed the opportunity to be the last barrier due to not being familiar with Technical Specification section 4, Table 4.1-3 sampling requirements.

V. Automatic or Manually Initiated Safety System Responses:

No safety system responses occurred or were required to occur.

VI. Assessment of the Safety Consequences and Implications of the Event:

The failure to obtain a sample of the spent fuel pool following its fill on January 23, 1998 had no adverse safety consequences.

Section 5.4.1 of TMI-1 Technical Specifications states "When fuel is being moved in or over the Spent Fuel Storage Pool "A" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition." It also states that "When fuel is being moved in or over the Spent Fuel Storage Pool "B" and fuel is being stored in the pool, a boron concentration of at least 600 ppmb must be maintained to meet the NRC maximum allowable reactivity value under the postulated accident condition. The bases of section 4.1 of the Technical Specifications states "The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb."

No movement of fuel was conducted between the time the spent fuel pool was filled on 1/23/98 and the next sample was taken on 1/28/98. If fuel movements had been planned, boron samples would have been taken in accordance with procedure 1505-1. The Technical Specifications Bases clearly indicate that no minimum boron concentration is needed in the spent fuel pool for safe plant operation, except during fuel movements. Because the boron concentration of the spent fuel pool is typically above 2500 ppmb (2897 ppmb following the fill) no normal filling operation (outside of filling because of a major leak in the pool, which was not on-going) could dilute the boron concentration significantly below its initial value.

During the review of this event, it was determined that the TMI-1 Technical Specification Surveillance requirement for Spent Fuel Pool water sampling is different than the Standard Technical Specification (STS) requirement, which is to verify boron concentration every 7 days. The STS bases for this surveillance frequency states: "the 7 day frequency is

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 Harris pools C and D. These findings are necessarily of an interim nature, because Orange County 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 for PWR fuel in Harris pools C and D are summarized, thereby showing the broad outlines of the envelope of criticality for these pools.

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 employed for probabilistic risk assessments (PRAs) at nuclear power plants. However, Orange County 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, Orange County 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).

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 this case suggests that no entity in the United States has undertaken the calculations necessary to determine the envelope of criticality in a fuel pool. 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 Orange County. 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.

Most 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 involved 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.[1]

In addition, where several fuel pools are interconnected but are separated by removable gates, as are the four pools at the Harris plant, water from one pool could mix with water from another pool if a gate is removed. If one pool has a lower concentration of soluble boron, the mixing process will reduce the concentration in the other pool. A similar effect could occur if a pool enters into communication with a fuel transfer canal or the reactor refuelling cavity.

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.

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

A study by the Westinghouse Corporation sought to estimate the probability of soluble boron dilution at PWR plants.[2] 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.[3] 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.

2.7 Criticality Calculations for Harris Pools C and D

In its application for a license amendment to activate pools C and D at Harris, CP&L provided the results of some calculations related to criticality.[4] These results were not sufficient to support an assessment of the probability or consequences of a criticality event in pool C or pool D. However, additional calculations have subsequently been performed by CP&L and the NRC Staff, and these show the broad outline of the envelope of criticality for pools C and D, for scenarios involving fuel mispositioning and the dilution of soluble boron.

The NRC Staff submitted a request for additional information (RAI) to CP&L on April 29, 1999. Question 1 of that RAI requested an analysis of a fuel mispositioning event in which one fresh PWR assembly is inappropriately placed in pool C or pool D at Harris. This placement would violate the burnup/enrichment restrictions which are specified in Figure 5.6.1 of the proposed new Harris Tech Specs.

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

3 Deposition of Dr. Laurence I. Kopp, Tr. at 36-39. A copy of the relevant pages of Dr. Kopp's deposition is attached as Exhibit C-1.

4 See Revision 3, Enclosure 7 to CP&L's license amendment application.

In its response of June 14, 1999 to the RAI, CP&L asserted that a souble boron concentration of 400 ppm would be sufficient to maintain Keffective less than 0.95 if this mispositioning event occurred. No supporting calculations were provided.

The results of some additional calculations relevant to the RAI were provided by CP&L in a letter of October 15, 1999, to which was attached a letter of October 11, 1999 from Holtec. These results were supported by a proprietary Holtec document which provided some details about the calculations. The proprietary document is not cited here.

For the mispositioning event postulated in the April 29, 1999 RAI, CP&L's additional calculations showed that Kinfinite would be 0.9916 (with a 95%/95% confidence level) in the absence of soluble boron.[5] These calculations assumed the placement of one fresh PWR fuel assembly (enriched 5 wt% in U-235) surrounded by PWR fuel of the maximum reactivity permitted by Figure 5.6.1 of the proposed new Tech Specs. CP&L also calculated that the maximum Kinfinite would be 0.9352 if the soluble boron concentration were 400 ppm. Further calculations showed a maximum Kinfinite of 0.8671 (0.7783) for a soluble boron concentration of 1,000 (2,000) ppm.

In a variant of its calculation that assumed an absence of soluble boron, CP&L assumed that the one fresh PWR assembly is placed in a PWR cell adjacent to the BWR storage racks. Assuming that this assembly is surrounded by PWR and BWR fuel of the maximum permitted reactivity, CP&L calculated that Kinfinite would be 0.9932 (with a 95%/95% confidence level).

Some related calculations were performed by the NRC Staff, 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

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[5] A fuel pool can contain a relatively large array of fuel. Thus, the difference between Keffective and Kinfinite will be relatively small for many pool situations. As a result, the approach to criticality in a fuel pool is often discussed in terms of the value of Kinfinite. The discussion in this appendix largely follows that practice.

[6] A copy of this document is provided herewith as Exhibit C-2.

an entire PWR rack of the type proposed for Harris pools C and D is loaded with fresh PWR fuel assemblies enriched 5 wt% in U-235.

The SCALE modular code system was used by the NRC Staff for these calculations, and the Ulses Memorandum compared the results of the SCALE calculations with the results of CP&L calculations. The Ulses 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, the SCALE calculations yielded a K of 1.19378. For the same problem, using the CASMO (MCNP) code, CP&L calculations were said by the Ulses memorandum to yield a K of 1.2076 (1.2056). These CP&L results appear to be the results presented for PWR racks in Table 4.5.1 of Revision 3 of Enclosure 7 to CP&L's license amendment application. In that table, the CASMO result is said to be Kinfinite, whereas the MCNP result is said to be Keffective. The MCNP result makes some relatively small allowances for uncertainty, bias and temperature variation.

The Ulses memorandum also provided the results of calculations for a problem in which a PWR rack in Harris pool C or D is loaded with PWR fuel burned to 41,700 MW-days per tonne U, without the presence of any soluble boron. SCALE calculations yielded a K of 0.8940, while CASMO calculations by CP&L were said to yield a K of 0.9126. This CASMO result appears to be the result presented in Table 4.2.1 of Revision 3 of Enclosure 7 to CP&L's license amendment application. In that table, a Kinfinite of 0.9126 is reported as a CASMO result before allowances are made for uncertainities and the effect of axial burnup distribution.

The above-presented results may be summarized in simple terms. Assuming an absence of soluble boron, consider three cases. First, a rack filled with well-burned (42,000 MW-days per tonne U) PWR fuel will be clearly subcritical, with a Kinfinite of about 0.9. Second, a rack filled with PWR fuel of the highest permissible reactivity, plus one fresh PWR assembly, will be close to criticality, with a Kinfinite of about 0.99. Third, a rack filled with fresh PWR fuel will be clearly supercritical, with a Kinfinite of about 1.2.

Now consider the presence of soluble boron in various concentrations, assuming a rack in which one fresh PWR fuel assembly is surrounded by PWR fuel of the highest permissible reactivity. A soluble boron concentration of 400 ppm will yield a Kinfinite of about 0.94, while a concentration of 1,000 ppm will yield a

Kinfinite of about 0.87 and a concentration of 2,000 ppm will yield a Kinfinite of about 0.78.

If these results are accepted, it follows that the envelope of criticality for PWR fuel in Harris pool C or D, for scenarios involving fuel mispositioning and soluble boron dilution, will involve the placement in a pool of two or more fuel assemblies with a reactivity that exceeds the permissible level. Also, it appears that the presence of soluble boron at a concentration of 2,000 ppm will preserve a subcritical margin of reactivity even if the racks are filled with fresh fuel. Thus, the envelope of criticality will be a set of circumstances which combine the mispositioning of two or more fuel assemblies with the presence of soluble boron in concentrations between zero and some level less than 2,000 ppm.

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 Harris pool C or D as a result of the mispositioning of PWR 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 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 CP&L and the NRC staff show that the final configuration could be critical on prompt neutrons alone. For example, CP&L finds that an almost-critical configuration exists (Kinfinite is 0.99) if one fresh PWR fuel assembly is present in a rack and soluble boron is absent. The completed placement of additional fresh assemblies in nearby locations 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.[7]

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 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

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

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 CP&L and the NRC staff for Harris pools C and D show that supercritical configurations could occur if two or more 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

Transcript of Deposition of Dr. Laurence I. Kopp,
Pages 35-40 (November 4, 1999)

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

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that's why we decided this week to actually do a calculation and see if would be true for Shearon Harris. And we found we are subcritical for the entire rack.

Q. Okay. Under what circumstances, if any, and under what regulatory requirements, if any, does the NRC require the reporting of errors in controlling boron concentration in the water of fuel storage pools?

A. I'm not sure if there would be any requirements for reporting that. If the boron concentration were a minimum boron concentration were in tec specs and if that were violated during the surveillance interval, there would be a certain amount of time where one could reborate and get back up to the required minimum level. And that would not be really I guess reportable unless one did not borate in time. There's a certain interval where you come back within regulations.

A. I see. And if you correct it with appropriate intervals it's not a reportable event; is that what you're saying?

A. Right.

Q. Okay. To the extent that boron dilution events are reported to the NRC, does the NRC keep any centralized record of boron dilution events that you know?

A. It would be the same as the LER's for fuel misplacements. There would be the LER's as far as I know. We don't compile them but they're available.

Q. Has the NRC performed or obtained any analysis or evaluation of nuclear power plant operator's experience with controlling boron concentrations in fuel storage pools?

A. Not that I know of.

MS. CURRAN: I'd like to ask the court reporter to mark as Exhibit 10 an October 25th, 1996 letter from Timothy E. Collins, Acting Chief, Reactor System Branch, Division of System Safety and Analysis, NRC, to Mr. Tom Green, Chairman Westinghouse Owner's Group. Subject: Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Special Fuel Rack Criticality Analysis Methodology.

Attached to this cover letter is a Safety Evaluation by the Office of Nuclear Reactor Regulation relating to Topical Report WCAP-14416-P.

(Whereupon, Exhibit Number 10 was marked for identification.)

Q. Dr. Kopp, are you familiar with this document?

A. Yes, I am.

Q. If you would turn to page 10 -- actually page 10 is a continuation of a discussion that starts on page

8, Section 3.7 entitled Soluble Boron Credit Methodology; isn't that correct?

A. Yes.

Q. If you look at the second full paragraph on page 10 of the SER, I'd like to ask you about a sentence that reads: "However, a boron dilution analysis will be performed for each plant requesting soluble boron credit to ensure that sufficient time is available to detect and mitigate the dilution before the 0.95 k effective design basis is exceeded and submitted to the NRC for review." In parentheses, "Ref, dot, 29."

Can you explain to me what is meant by this sentence and the reference to Ref 29?

A. Yes. This is the new methodology that I spoke of earlier. This is one of the reasons for updating the Grimes letter. This is a recent approval we gave for crediting partial soluble boron in spent fuel pools. And since we are allowing, not for Shearon Harris, but for some reactors, credit for soluble boron under normal conditions to meet .95, this would now require a new accident to be evaluated which would be the boron dilution event.

For other plants, such as Shearon Harris, which do not take credit for soluble boron during normal conditions, the fact that they calculate the five percent

2 subcriticality margin in pure water takes care of the boron dilution event, that is complete dilution.

For these newer plants that want to take credit for the new methodology. They still must show they are subcritical with no boron, k effective is less than one, but to meet the k arc criteria, k effective less than or equal to .95, they can take credit for a certain amount of soluble boron. So because of that we require them now to do a boron solution analysis to show that they would get them below .95 dilution event.

Q. Okay. But Reference 29 in parentheses, when I turn to the back of this SER, Reference 29 is "Cassidy, B., et. al., Westinghouse Owners Group Evaluation of the Potential for Diluting PWR Spent Fuel Pools, WCAP-14181, July 1995."

How does that Reference 29 relate to what we were just reading on page 10?

A. That was a companion to this Westinghouse report which requested credit for partial boron. In order to prove that methodology I said they have to do a boron dilution event analysis. And this other report that you referenced shows how to do an analysis of a boron dilution event in the PWR.

Q. So the reason for the mention of Reference 29 is that this is a way for licensees to do the boron dilution analysis and that, that will meet NRC approval?

A. When they want credit for this methodology, partial boron credit, yes.

Q. And has the NRC approved Reference 29 for that purpose?

A. No. The approval of a boron dilution event we decide is done on a case by case basis because the plans vary so much. The amount of, the volume of water that can be inserted into a pool for dilution varies from plant to plant through the mode of inserting it, the capacity of the pools vary. We decided a generic dilution event would not be worth anything or worth much, so we decided to, the people that wanted to accept this methodology for partial boron credit would have to do a plan specific for boron dilution analysis for their specific spent fuel pool. That's why that boron dilution event was never approved or accepted. It was a generic type of topical report.

Q. Okay.

Q. Has the NRC performed or obtained any analysis of the probability and/or consequences of potential accidents resulting from improper boron concentration in fuel storage pool water?

A. Only the analysis that shows that the zero PPM of boron when there's still a five-percent subcritical

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:

ERRATA SHEET

PLEASE ATTACH TO THE DEPOSITION.

IN THE CASE OF : Carolina Power & Light
CASE # : 99-762-02 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 "comparicances" to "concurrences"	"
29	6	change "wasn't" to "was"	"
29	12	change "200" to "with 2000"	"
29	15	change "pool" to "fuel"	"
38	6	change "arc" to "effective"	"
38	9	change "solution" to "dilution"	"
39	7	change "plans" to "plants"	"
39	15	change "plan" to "plant"	"

EXHIBIT C-2

Memorandum from Tony P. Ulses, NRC,
To Ralph Caruso, NRC re: Completion of Criticality
Assessment of Misloading Error in Harris C and D
Spent fuel Pool (November 5, 1999)

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:

<u>Typical Parameters</u>

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

<u>Problem 1</u>

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.

<u>Problem 2</u>

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 output for the global unit from KENO-VI in Appendix A and prepared several spectral plots. The information from the major edit indicates that all of the parts of the problem have been sampled. Note that the flux for region 1 in the global unit is zero because region 1 represents the hole containing the fuel which was inserted into the global unit. The flux should be zero in the global unit for this region.

The spectral plots are presented as Figures 1 and 2. The error bars represent one standard deviation and were extracted from the major edit (see Appendix A). From these plots we can ascertain that there are no unexpected trends in the results. For example, figure 1 shows a characteristic light water moderated reactor spectrum, but the thermal peak is smaller than it would be in the reactor. This reduction is caused by the additional absorption in the rack poison. Furthermore, we can see that we had complete coverage of the energy domain and that the sampling was significant enough to reduce the standard deviation to acceptable values.

5 Conclusions

Analyses have been performed to assess the effect of the worst case misloading scenario in the Harris "C" and "D" spent fuel pool. This analysis demonstrates that the maximum possible multiplication factor in the "C" and "D" spent fuel pools is 0.98 assuming that one credits the soluble boron present in the pool coolant. It should be noted that this analysis does not consider manufacturing tolerances, but the multiplication factor bias from manufacturing uncertainties is typically not larger than 1%. The staff has also been able to confirm that the methods used by the licensee contractor yield results that are consistent with the staff's results.

6 References

1. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998. (Holtec International Proprietary)

2. "SCALE 4.3, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Oak Ridge National Laboratory, 1995.

3. M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.

4. O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

5. W. C. Jordan, et. al., "Validation of KENOV.a, Comparison with Critical Experiments" ORNL/CSD/TM-238, Oak Ridge National Laboratory, 1986.

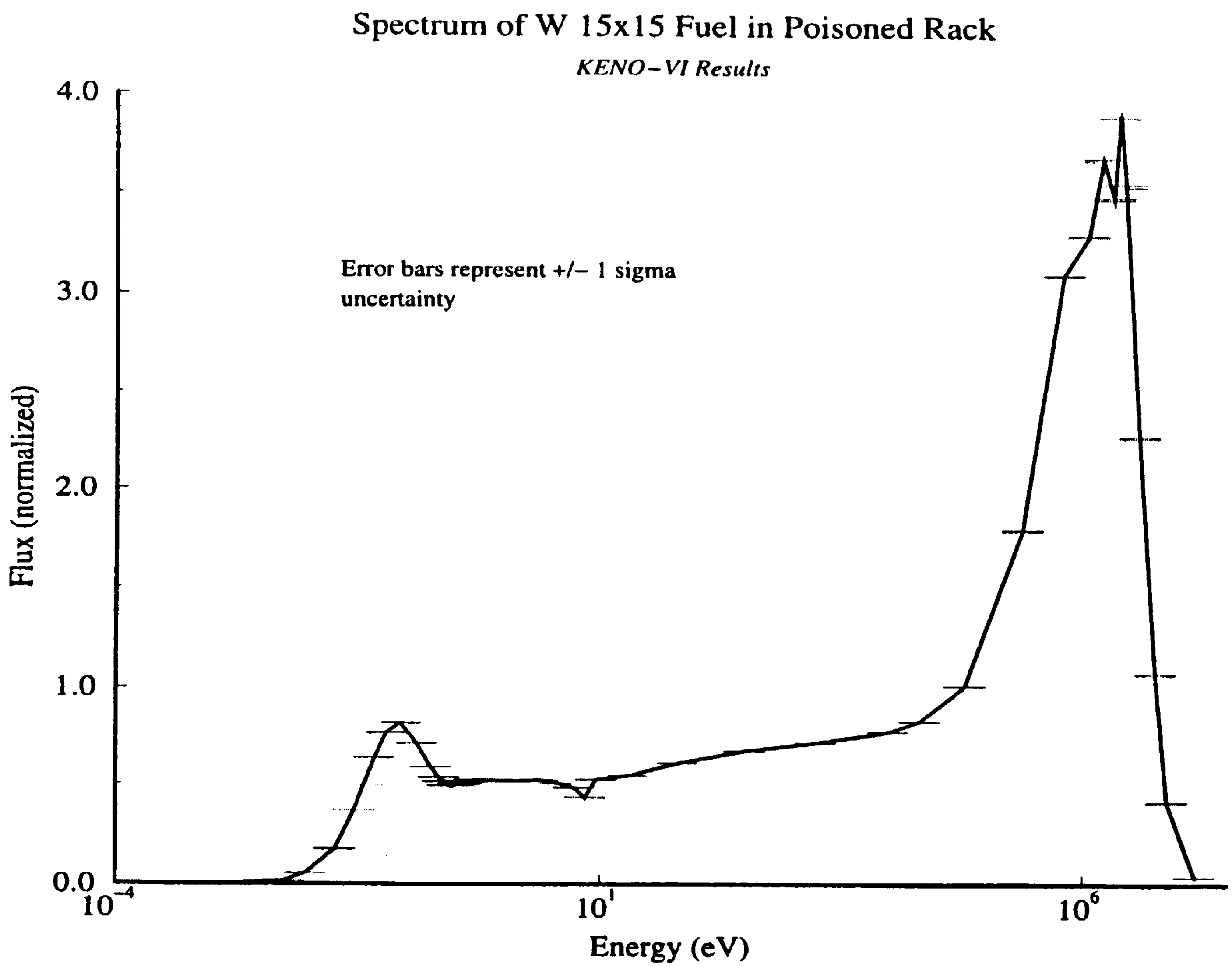

Figure 1 KENO Predicted Spectrum for W 15x15 Fuel Assembly

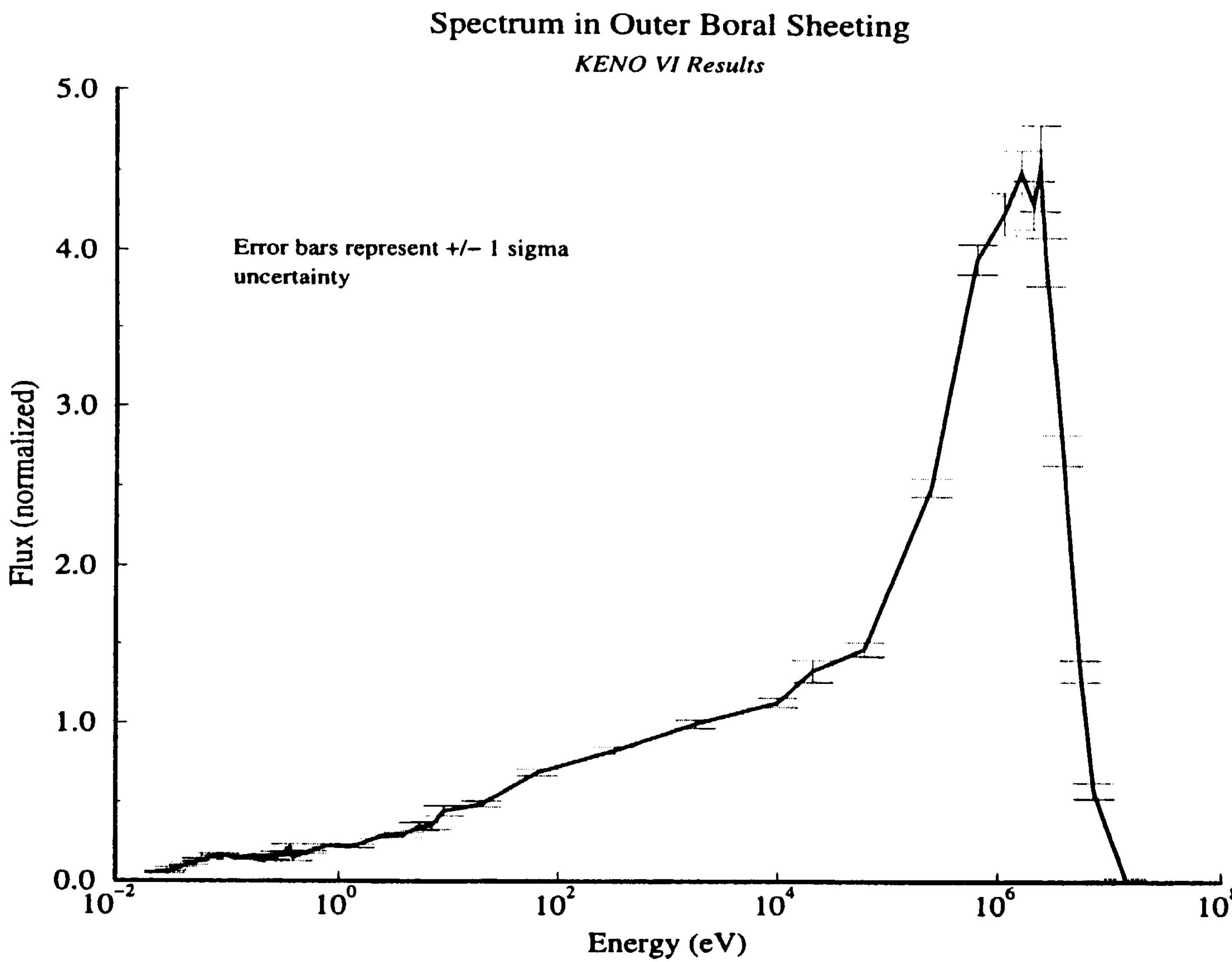

Figure 2 KENO Predicted Spectrum in Outer Boral Sheeting

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

LIST OF EXHIBITS TO ORANGE COUNTY'S SUMMARY AND SWORN SUBMISSION REGARDING CONTENTION TC-2

1. Declaration of Dr. Gordon Thompson in Support of Orange County's Summary and Sworn Statement Regarding Contention TC-2 (January 4, 2000)

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. Letter from Donna B. Alexander, CP&L, to U.S. NRC, enclosing response to April 29, 199, RAI (June 14, 1999)

6. Transcript of Deposition of Michael J. DeVoe, P.E. (October 20, 1999)

7. AEC Press Release entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits" (November 22, 1965)

8. 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)

9. 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)

10. Letter from J. J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and attached draft of General Design Criteria (relevant excerpts)

11. 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)

12. Notice of proposed rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

13. Letter from William B. Cottrell, ORNL, to H. L. Price, AEC (September 6, 1967) ane enclosed ORNL comments on proposed GDC.

14. 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) (relevant excerpts)

15. 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)

16. Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

17. Letter from Donna B. Alexander, CP&L, to U.S. NRC (October 15, 199), enclosing letter from Scott H. Pellet, Holtec International, to Steven Edwards, CP&L (October 11, 1999)

CONTENTION TC-2: EXHIBIT 1

Declaration of Dr. Gordon Thompson in Support of Orange County's Summary and Sworn Statement Regarding Contention TC-2 (January 4, 2000)