UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON TECHNICAL CONTENTIONS 2 AND 3

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.1113, the Nuclear Regulatory Commission staff (Staff) hereby submits its written presentation summarizing all the facts, data and arguments of which the Staff is aware and on which the Staff intends to rely at oral argument, scheduled for January 21, 2000. For the reasons set forth below, the Staff submits that there is no genuine and substantial dispute of fact or law relating to the Board of Commissioners of Orange County's (BCOC) Technical Contentions 2 and 3. This written summary is supported by the affidavits of Richard Laufer, Laurence C. Kopp, Anthony P. Ulses, Kenneth Heck, Donald Naujock, James Davis and Christopher Gratton.

II. BACKGROUND

On December 23, 1998, Carolina Power & Light Company (Licensee or CP&L) filed an application for a license amendment, pursuant to 10 C.F.R. § 50.90, for the Shearon Harris Nuclear Power Plant (Harris or HNP). Letter to United States Nuclear Regulatory Commission from James Scarola, Vice President, Harris Nuclear Plant, Carolina Power & Light Co., December 23, 1998 (Application). *See* Laufer Affidavit, Exhibit). The Application was submitted under oath or affirmation, as required by 10 C.F.R. § 50.30(b), and sought approval to increase spent fuel storage capacity by adding rack modules to two spent fuel pools ("C" and "D") and placing the two pools into service. Laufer aff., Ex. 1 at 1. On January 13, 1999, the NRC published a notice of proposed no significant hazards consideration determination and opportunity for hearing.[1] On February 12, 1999, BCOC filed a request for hearing and petition to intervene.[2] The petition to intervene was granted by the Atomic Safety and Licensing Board (Board) by Memorandum and Order (Order), dated July 12, 1999. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-99-25, 50 NRC 25 (1999). The Board admitted two of BCOC's technical contentions for litigation. *Id.* at 38.

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[1] "Carolina Power & Light; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination and Opportunity for a Hearing," 64 Fed. Reg. 2237 (1999).

[2] Orange County's Request for Hearing and Petition to Intervene, February 12, 1999.

On July 21, 1999, pursuant to 10 C.F.R. § 2.1109, the Licensee filed a request to invoke the hybrid hearing procedures of Subpart K and for oral argument.[3] On July 29, 1999, the Board granted the request and established a discovery schedule, a schedule for filing written presentations and a date for oral argument.[4] On December 1999, the Board issued an Order granting the Staff's oral motion to extend the time to file written presentations and the date for oral argument.

In accordance with the Board's Orders and 10 C.F.R. § 2.1113, the Staff hereby submits its written summary of all the facts, data, and arguments known to the Staff and on which the Staff intends to rely at oral argument to refute the existence of a genuine and substantial dispute of fact as to Technical Contentions 2 and 3.

III. STATEMENT OF FACTS

Shearon Harris Nuclear Power Plant, Unit 1 (HNP) is a three-loop Westinghouse pressurized water reactor (PWR) operated by CP&L in Wake and Chatham Counties, North Carolina.

The NRC issued the construction permit for HNP, which was originally planned as a four nuclear unit site, on January 27, 1978. (Laufer aff. at ¶ 3). In order to accommodate four units at HNP, the fuel handling building (FHB) was designed and constructed with four

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[3] Applicant's Request for Oral Argument to Invoke Subpart K Hybrid Hearing Procedures and Proposed Schedule, July 21, 1999.

[4] *Memorandum and Order (Granting Request to Invoke 10 C.F.R. Part 2, Subpart K Procedures and Establishing Schedule)*, July 29, 1999 (Subpart K Order).

separate pools capable of storing spent fuel. (Laufer aff. at ¶ 3). The two pools at the south end of the FHB, now known as spent fuel pools (SFPs) A and B were to support HNP Units 1 and 4. The two pools at the north end of the building, SPFs C and D, were to support HNP Units 2 and 3. The pools were designed to store both PWR and boiling water reactor (BWR) fuel. The multi-unit design included a SFP cooling and cleanup system to service SFPs A and B, which would be cooled by the Unit 1 component cooling water system (CCWS); and a separate cooling and cleanup system to support SFPs C and D which would be cooled by the Unit 2 CCWS. (Laufer aff. at ¶ 3).

HNP Units 3 and 4 were canceled in late 1981 and HNP Unit 2 was canceled in late 1983. (Laufer aff., Exhibits 3 and 4) The construction permits for Units 2, 3, and 4, expired on , respecitely, and were not renewed. (Laufer aff. at ¶ 4) The FHB, all four SFPs (including liners), and the cooling and cleanup system to support SFPs A and B were completed. (Laufer aff. ¶ 4) However, the construction on the SFP cooling and cleanup system for SFPs C and D was not completed. (Id.).

The NRC issued Facility Operating License No. NPF-63 for the full power operation of HNP Unit 1 on January 12, 1987. HNP's current FSAR prohibits the use of SFPs C and D until they are completed and made operational. (Laufer aff. at ¶ 5). The license authorized CP&L to receive and store spent fuel from its other nuclear plants (Brunswick Units 1 and 2, and H. B. Robinson, Unit 2) at HNP. Specifically, License condition 2.B.(8) states:

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

> (8) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, Carolina Power & Light Company to receive and posses but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Brunswick Steam Electric Plant, Units 1 and 2, and H. B. Robinson Steam Electric Plant, Unit 2.

As permitted by its Operating License, CP&L has implemented a spent fuel shipping program. Spent fuel from Brunswick (2 boiling water reactor (BWR) units) and Robinson (1 PWR unit) is shipped to HNP for storage in SFPs A and B. CP&L ships fuel to HNP in order to maintain full core offload capability at Brunswick and Robinson. As a result of the operation of HNP, shipping program requirements, and the unavailability of a Department of Energy (DOE) storage facility, CP&L has determined that it would be necessary to activate SFPs C and D and the associated cooling and cleanup systems by early in the year 2000. Activation of these two pools will provide storage capacity for all four CP&L nuclear units through the end of their current licenses. (Laufer aff., Exhibit 1).

On December 23, 1998, CP&L submitted a license amendment request to support placing SFPs C and D in service. (Laufer aff., Exhibit 1) The proposed action consists of three parts:

1. A revision to Technical Specification (TS) 5.6 to identify pressurized water reactor (PWR) burnup restrictions, boiling water reactor (BWR) enrichment limits, pool capacities, heat load limitations and nominal center-to-center distances between fuel assemblies in the racks to be installed in SFPs 'C' and 'D.'

CP&L proposed to use higher density fuel racks in SFPs C and D than are currently used in SFPs A and B. The use of the higher density racks requires additional administrative controls on PWR burnup and BWR enrichment to ensure Keff less than or equal to 0.95.

2. An alternative plan in accordance with the requirements of 10 CFR 50.55a to demonstrate an acceptable level of quality and safety in completion of the component cooling water (CCW) and SFPs 'C' and 'D' cooling and cleanup system piping.

In order to activate SFPs C and D, it is necessary to complete construction of the cooling and cleanup system for these pools and to install tie-ins to the existing HNP Unit 1 CCWS to provide heat removal capabilities. Approximately 80% of the SFP cooling and cleanup system piping and the majority of the CCW piping was installed during the original plant construction. At the time that construction on the SFP cooling system was discontinued following cancellation of HNP Unit 2, a formal turnover of the partial system was not performed and CP&L has since discontinued its N Certificate program. Also, some of the field installation records for the completed piping are no longer available. As a result, the system when completed will not satisfy ASME Section III code requirements (i.e., will not be N stamped). Therefore, CP&L submitted an Alternative Plan in accordance with 10 CFR 50.55a(a)3 to demonstrate that the completed system will provided an acceptable level of quality and safety.

3. An unreviewed safety question for additional heat load on the CCW system, which is not in contention in this proceeding

The proposed amendment was noticed in the Federal Register on January 13, 1998.

III. THE REGULATORY FRAMEWORK

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

Pursuant to the Board's Subpart K Order of July 29, 1999, this proceeding is governed by the hybrid hearing procedures of 10 C.F.R. § 2.1101, et seq. (Subpart K). Subpart K provides that its procedures may be used, at the request of any party, in contested proceedings concerning, *inter alia,* applications for a license amendment "to expand the spent fuel capacity at the site of a civilian nuclear power plant, through the use of high density fuel storage racks . . . ." 10 C.F.R. § 2.1103. The procedures include a 90 day discovery period, followed by submittal of a detailed written summary, and fifteen days thereafter, oral argument. 10 C.F.R. §§ 2.1111, 2.1113. The detailed written summary must contain all the facts, data, and arguments known to the party and on which the party intends to rely at oral argument to support or refute the existence of a genuine and substantial dispute of fact. 10 C.F.R. § 2.1113(a). All supporting facts and data must be submitted in the form of sworn written testimony or other sworn written submission. *Id.* The written submittals are to be simultaneously served on all other parties. *Id.*

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

Subpart K provides for a form of summary disposition procedure. 50 Fed. Reg. 41662, 41664 (1985). There are several differences between the provisions of Subpart K and traditional NRC summary disposition practice, including: simultaneous filing of pleadings; mandatory oral argument; and placing the burden of demonstrating the *existence* of a genuine and substantial issue of material fact is on the party requesting adjudication. *Id.* at 41667. *Compare Georgia Power Co.* (Vogtle Generating Plant, Units 1 and 2), ALAB-872, 26 NRC, 127, (1987).

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

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

50 Fed. Reg. at 41666-67.

B. Technical Contention 2 - Criticality

In Contention 2, the intervenor asserts that criticality prevention for pools C and D is inadequate, as proposed in the Application. The asserted bases for Contention 2 are that (1) GDC 62 prohibits the use of administrative measures, and CP&L proposes to use credit for burnup, an administrative measure, to prevent criticality in pools C and D and (2) one failure, misplacement of a fuel assembly, could cause criticality if credit for burnup is used.

In the Application, CP&L requested approval to place spent fuel pools C and D at Shearon Harris in service. Specifically, CP&L proposes to increase the spent fuel storage capacity by adding storage racks to pools C and D. With respect to criticality, the NRC staff reviewed the Application to determine if it satisfied the requirements of 10 C.F.R. Part 50, Appendix A, Criterion 62 . General Design Criterion 62 provides:

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

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

The "double contingency principle" of draft Regulatory Guide 1.13, revision 2, provides the analytical foundation for the Staff's analysis of criticality in spent fuel pools. That principle, as implemented by the Staff, is articulated in 10 C.F.R. § 72.124, which states:

> Spent fuel handling, packaging transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential

> changes have occurred in the conditions essential to nuclear criticality safety.

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

C. Technical Contention 3 - Quality Assurance

CP&L submitted a request for authorization to use an alternative plan to the ASME code requirements, pursuant to 10 C.F.R. § 50.55a(a)(3), for certain code required documents which are unavailable for ASME Code Class 3 piping and welds in the SFP cooling system. 10 C.F.R. § 50.55a renders each construction permit subject to the following requirements:

> (A)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

The Staff reviewed the Licensee's request against the criteria of 10 C.F.R. § 50.55a(a)(3)(i), which provides that:

> (3) Proposed alternatives to the requirements of . . . this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate:
>
> (i) The proposed alternative would provide an acceptable level of quality and safety....

Technical Contention 3 alleges that CP&L's proposal to provide cooling of pools C & D fails to satisfy the quality assurance criteria of 10 C.F.R. Part 50, Appendix B, specifically Criterion XIII, Criterion XVI, and Criterion XVII.

Appendix B requires the development and application of a quality assurance (QA) program for the design, fabrication, construction, and testing of the structures, systems, and components of the facility at the construction permit stage, and a QA program for manegerial and administrative controls at the operating license stage. Appendix B establishes the QA requirements for such structures, systems and components.

Criterion XIII provides, as pertinent here, that "[m]easures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration."

Criterion XVI provides that "[m]easures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. . . .

Criterion XVII provides that "[s]ufficient records shall be maintained to furnish evidence of activities affecting quality. . . . Records shall be identifiable and retrievable.

IV. THE ADMITTED CONTENTIONS

In its July 12, 1999 Contention Order, the Board admitted two technical contentions for litigation in this proceeding. In admitting the contentions, the Board stated that they were "accepted for litigation in the form and subject to the interpretations set forth [in the Order]. *Harris*, LBP-99-25, 50 NRC at 38.

The first of the two contentions, designated TC-2, concerns criticality, and states:

> 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.

Id. at 35. The Board admitted the contention with two bases. The first basis, as admitted by the Board, states: "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." *Id.* The Board designated this a question of law and limited its admission to "legal arguments on whether the use of administrative limits on burnup and enrichment of fuel stored in pools C and D properly conforms to the requirement s of GDC 62 for the prevention of criticality." *Id.* at 36.

Basis 2, as admitted by the Board, reads: "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." *Id.* The Board designated this basis as 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?" *Id.* The Board went on to say that further inquiry on the validity of calculations involved in determining criticality is warranted "in determining whether the required single failure criterion is met." *Id.*

The remaining contention, TC-3, concerns quality assurance., and reads:

> CP&L's proposal to provide cooling of pools C & D by relying upon the use of previously completed portions of the Unit 2 Fuel Pool Cooling and Cleanup System and the Unit 2 Component Cooling Water System fails to satisfy the quality assurance criteria of 10 C.F.R. Part 50, Appendix B, specifically Criterion XIII (failure to show that piping and equipment have been stored and preserved in a manner that prevents damage or deterioration), Criterion XVI (failure to institute measures to correct any damage or deterioration), and Criterion XVII (failure to maintain necessary records to show that all quality assurance requirements are satisfied).
>
> Moreover, the Alternate Plan submitted by Applicant fails to satisfy the requirements of 10 C.F.R. § 50.55a for an exception to the quality assurance criteria because it does not describe any program for maintaining the idle piping in good condition over the intervening years between construction [and] implementation of the proposed license amendment, nor does it describe a program for identifying and remediating potential corrosion and fouling.
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> The Alternative Plan submitted by Applicant is also deficient because 15 welds for which certain quality assurance records are missing are embedded in concrete and inspection of the welds to demonstrate weld quality cannot be adequately accomplished with a remote camera.
>
> Finally, the Alternative Plan submitted by Applicant is deficient because not all other welds embedded in concrete will be inspected by remote camera, and the weld quality cannot be demonstrated adequately by circumstantial evidence.

Id. at 36-37. In admitting the contention, the Board delineated five issues: 1) whether the criteria of Appendix B are to be enforced; 2) whether the criteria of 10 C.F.R. § 50.55a(a)(3) will be met; 3) if the criteria of 10 C.F.R. § 55.55a(a)(3)(i) (an acceptable level of quality and safety) is the basis of review, whether a failure of quality control could lead to a hazard; 4) whether "proper" storage and maintenance during the time between construction and implementation of the license amendment is necessary as a matter of law and fact; and 5) what is the scope of any failure to meet the requirements of the regulations.

V. ARGUMENT

A. TECHNICAL CONTENTION 2, CRITICALITY CONTROL

1. BCOC's Designated Expert Witness, Gordon Thompson, Should Be Disqualified As an Expert Witness and his Testimony/Declaration Stricken

BCOC proffered Gordon Thompson as its expert witness for Contention 2, inadequate criticality control. The staff submits that BCOC has not demonstrated Dr. Thompson's expertise in criticality control or any issue related to Contention 2.

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

While the Federal Rules of Evidence are not directly applicable to Commission proceedings, NRC adjudicatory boards often look to those rules for guidance, and have adopted the standard for expert witnesses enunciated in Rule 702 as allowing a witness to be qualified as an expert by "knowledge, skill, experience, training, or education" to testify "[i]f scientific, technical, or other specialized knowledge will assist the tried of fact to understand the evidence or to determine a fact in issue." *McGuire*, ALAB-669, 15 NRC at 475; *Southern California Edison Co.* (San Onofre Nuclear Generating Station, Units 2 and 3), ALAB-717, 17 NRC 346, 365 n.32 (1983); *Philadelphia Elec. Co.* (Peach Bottom Atomic Power Station Units 2 and 3), ALAB-701, 10 NRC 1517, 1524 (1982).

In the *McGuire* case, proposed expert testimony on hydrogen combustion and control was excluded as non-expert, where the witness, a physical organic chemist with a master's degree, claimed to have an "ability to understand and evaluate matters of a technical nature," based on his "academic and practical training" and "years of reading AEC and NRC documents," but lacked specific expertise in the subject in issue. *McGuire*, ALAB-669, 15 NRC 453. *See also Philadelphia Elec. Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 687, 733 (1985) (Art therapist, with no background relating to the issue, not qualified to give expert testimony); *Pacific Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), LBP-87-25, 26 NRC 168, 182 (1987) (Testimony of Ph.D. in physics with no experience with the technical subjects under review: nuclear engineering, nuclear systems, nuclear criticality, seismic design, etc., admitted but only given the weight appropriate considering his Ph.D. and years of teaching physics); *Philadelphia Elec. Co.* (Peach Bottom Atomic Power Station Units 2 and 3), ALAB-701, 16 NRC 1517, 1523-24 (1982) (Ph.D. in chemistry not qualified to give expert opinion on health effects of radon releases, due to lack of education or experience in medicine, health physics or other areas related to health effects of radon). *Cf. Florida Power & Light Co.* (Turkey Point Nuclear Generating Plant, Units 3 and 4), LBP-86-23, 24 NRC 108, (1986) (Testimony of Ph.D. in mathematics with no training or knowledge of engineering, heat transfer and other technical issues under review, permitted to testify and act as expert interrogator to "sharpen the issues").

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

During his deposition, portions of which are attached as Brief Exhibit 1,[5] Dr. Thompson made the following claims regarding his proffered expertise:

> 1) He was hired by BCOC to provide technical and safety advice to support its intervention in this matter (Dep. Tr. 14-15);
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> 2) He has a Ph.D. in applied mathematics (Dep. Tr. 21);
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> 3) He has no training in fission reactor engineering, fission reactor criticality control or fission reactor criticality analysis (Dep. Tr. 21);

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[5] For convenience, the Staff will denote citations to Dr. Thompson's deposition transcript as "Dep. Tr.," rather than "Brief Exhibit 1."

4) He claims to be an expert in fission reactor criticality analysis for the purpose of this proceeding based on his "basic expertise in scientific principles and analytic principles" and his "general experience with engineering in general and nuclear plant engineering in specifics." His contribution to this proceeding will rely upon the application of general scientific principles to the criticality contention (Dep. Tr. 21-22);

5) He has no training in running criticality analysis codes and will not be running any codes in connection with this proceeding, (Dep. Tr. 23). He will confine his analysis to the assessment of the sufficiency of the assumptions which go into the analysis as to whether they address the issues of concern in connection with criticality. He will not address the analysis itself, because he is not competent to do so (Dep. Tr. 24-25);

6) He has never been licensed as a nuclear power plant operator. He has no training or experience in the operation of a nuclear power plant, engineering at a nuclear power plant, or writing or implementing procedures at a nuclear power plant (Dep. Tr. 25-26);

7) He is not an expert in nuclear power plant operations, but he claims to have "performed studies and presented testimony relating to the safety of nuclear facilities, including nuclear power plants; and in the course of those studies and preparing those testimonies, [he] has become expert in operational matters pertinent to the analyses and testimony. So in that limited sense, [he] is an expert in operations. It's [sic] a very circumscribed sense." (Dep. Tr. 26-27);

8) He claims to be familiar in a general sense with the configuration of the fuel building and the equipment at the Harris plant and the fuel management procedures, based upon his review of the FSAR and other CP&L documents and additional information gained from sitting through the deposition of Michael DeVoe, a CP&L employee. He is not

familiar with all the fuel management procedures, nor has he applied any of the procedures (Dep. Tr. 27-29);

10) He claims to be familiar with details of several nuclear facilities in several countries. He claims to have "always taken pains to acquire the necessary familiarity with the details of the design and operation of each facility in order to support whatever claim [he] made" (Dep. Tr. 30-31);

11) He spent approximately one hour in the fuel handling building at Harris. (Dep. Tr. 33) He has also been in several other fuel handling buildings for similar periods of time (Dep. Tr. 34-36);

12) None of the other projects he worked on, his publications, or his expert presentations and testimony have dealt with his analysis of assumptions used in criticality analysis (Dep. Tr. 38-39).

Based on the sworn testimony of Dr. Thompson during his deposition, as set forth above, it is clear that he lacks expertise related to an issue relevant to Contention 2. Moreover, by his own admission, none of the numerous "Project sponsors and tasks," "Publications," or "Expert testimony" listed in Dr. Thompson's Curriculum Vitae relate to spent fuel pool criticality issues. Dep. Tr. at 38-39. His Curriculum Vitae, therefore, does not demonstrate that Dr. Thompson has any knowledge, skill, experience, training or education related to the subject matter of Contention 2 - criticality control. The Curriculum Vitae gives no indication of the issues Dr. Thompson addressed in these projects or his area of expertise applied to them. Nor was the outcome of any project or expert testimony indicated. In addition, none of his listed publications, other than that related to this proceeding, concern any of the issues, data or facts involved herein.

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

As demonstrated above, Dr. Thompson does not qualify as an expert witness by virtue of his knowledge, skill, experience, training, or education. He is no more qualified to render an expert technical opinion on criticality than any other layperson. Therefore, any conclusions he makes, opinions he renders, or other testimony related to this contention must be stricken.[6]

2. Statement of Facts

The facts upon which the Staff relies with respect to Contention 2 are set forth in the affidavits of Dr. Laurence I. Kopp (Kopp Affidavit) and Anthony P. Ulses (Ulses Affidavit). Dr. Kopp's and Mr. Ulses' professional qualifications are summarized in their affidavits and

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[6] Since the intervenor is not offering Dr. Thompson as a fact witness, the Board cannot merely limit his testimony to facts based on personal observation.

set forth in detail in attachments to the affidavits. Both Dr. Kopp and Mr. Ulses are qualified as experts on criticality by virtue of their education, experience, and demonstrated knowledge and skill regarding that subject. Kopp Affidavit, ¶ 1; Ulses Affidavit, ¶ 1. Dr. Kopp has been the principal criticality reviewer for most of the plants that have obtained NRC approval for the use of burnup credit for spent fuel storage. Kopp Affidavit, ¶ 19. Mr. Ulses has used the SCALE package of physics codes to perform reactor physics simulations to support Staff safety evaluations since early 1997. Ulses Affidavit, ¶ 1. A summary of the facts presented in the affidavits is set forth below.

Criticality is the achievement of a self-sustaining nuclear chain reaction. Kopp Affidavit, ¶ 7. The chain reaction proceeds as atoms of a fissile material absorb slow (thermal) neutrons and split (fission) into new light atoms (*i.e.*, fission products) and additional neutrons that, in turn, interact with additional fissile atoms. *Id.* Neutrons resulting from fission have high energy and are called "fast" neutrons. *Id.* Fast neutrons are not readily captured in U-235, the fissile material originally present in fresh fuel. Rather, a neutron must lose energy and "slow down," or become "thermalized" (a thermal neutron), in order to be readily captured in U-235 and cause fission. *Id.*

In order for fast neutrons to slow down, they must collide with, and transfer energy to, atoms. *Id.*, ¶ 8. This process is called "moderation." *Id.* A light element (such as hydrogen) is an effective moderator because the mass of its nucleus is on the same order as that of a neutron. *Id.* Therefore, upon initial collision, the neutron imparts most of its energy to the hydrogen nucleus and becomes thermalized. *Id.* Water, with its high hydrogen content, is the moderator in a light water reactor (LWR) such as Harris. *Id.*

After being created through fission, during the process of moderation, and after reaching thermal energy levels, a neutron may undergo several events. *Id.*, ¶ 9. It may be absorbed by nonproductive capture in the fuel, the moderator, or the structural materials. *Id.* It may leak from the reactor system and either be reflected back into the system or be lost. *Id.* Finally, it may be absorbed by the U-235, cause fission, and produce more fast neutrons. *Id.*

When the process continues on its own, the system of atoms of fissile material is said to be critical. *Id.*, ¶ 10. The measure of criticality is the effective neutron multiplication factor, k-effective, or k_{eff}. *Id.* The multiplication factor is the ratio of the rate of neutron production to neutron loss due to fission, nonproductive capture, and leakage. *Id.* K-infinity, or k_{inf}, is the infinite multiplication factor, which refers to the neutron muliplication of an infinite system. Ulses Affidavit, ¶ 7. For a given system or array of fuel, k_{inf} is always greater than k_{eff} because k_{inf} does not include loss of neutrons from leakage. *Id.* Criticality is achieved when k_{eff} is equal to 1.0. Kopp Affidavit, ¶ 10. When k_{eff} is less than 1.0, the system is subcritical. *Id.* Criticality can only occur in an array of LWR fuel if sufficient fissile material is available in a near-optimum geometry and a moderator (water) is present. *Id.* No array of LWR fuel can achieve criticality without water moderation present in the array. *Id.* Well-developed mathematical models (equations) exist in present-day computer codes and are used to compute k_{eff}. *Id.*

"Reactivity" is defined as $(k_{eff}-1)/k_{eff}$. *Id.*, ¶ 11. When fuel is irradiated in a reactor as a result of operation and power generation, the reactivity of the fuel decreases over the design life of the fuel assembly. *Id.* This reduction of reactivity with irradiation is called "burnup." *Id.* Burnup is caused by the change in fissile content of the fuel (*i.e.*, depletion of U-235 and production of Pu-239 and other fissile actinides), the production of actinide absorbers, and the production of fission product neutron absorbers. *Id.* Before each reactor operating cycle, a licensee performs a reload analysis that predicts the burnup of each fuel assembly during the cycle. *Id.* These calculations are confirmed during the cycle by measurements of various operating characteristics, such as boron concentration and power distribution. *Id.* After every operating cycle (typically 1 to 2 years), approximately 1/3 of the fuel in a reactor is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment. *Id.* This irradiated (or spent) fuel is generally placed in a spent fuel pool at the reactor site and is replaced in the reactor by fresh (unirradiated) fuel. *Id.*

General Design Criterion (GDC) 62 of 10 C.F.R. Part 50, Appendix A, requires that licensees prevent criticality in their spent fuel pools. *Id.*, ¶ 12. The NRC has established a 5% subcriticality margin for wet storage of spent fuel assemblies to assure that licensees meet the requirements of GDC 62. *Id.*, ¶ 14. Burnup credit is the practice of accounting for the reduced reactivity of spent fuel due to fissile material decay and fission product buildup described above in evaluating criticality safety. *Id.*, ¶ 13. Burnup of the fuel occurs as a natural consequence of the fuel being used in the reactor. *Id.* Therefore, fuel burnup is a physical process. *Id.* The regulations do not prohibit the use of burnup credit for criticality safety. *Id.*

CP&L proposes to use administrative procedures at Harris to verify that a fuel assembly has achieved the required amount of burnup to be placed in the Pool C or D storage racks. *Id.*, ¶ 15. CP&L is currently licensed to store fuel from two other CP&L plants, H. B. Robinson Steam Electric Plant, Unit 2 (Robinson), and Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick), as well as fuel from the Harris reactor core, in existing spent fuel pools A and B at Harris. *Id.* CP&L has stated that it selects spent fuel assemblies for shipment to Harris from Robinson and Brunswick in accordance with plant procedure NFP-NGGC-0003. *Id.* The purpose of this procedure is to assure that the selection of spent fuel to be shipped to Harris is acceptable for transportation and storage in the Harris A and B spent fuel pools. *Id.*

CP&L uses a computer program in conjunction with this fuel selection procedure. *Id.*, ¶ 16. For candidate assemblies to be shipped, the program retrieves the fuel type, enrichment, and burn-up from a database. *Id.* The fuel type and initial enrichment data for each fuel assembly contained in this database is based on manufacturing records. *Id.* The burnup data for each fuel assembly included in this database is obtained from the reload core design calculations and confirmed by periodic core monitoring of boron concentration and power distribution. *Id.*

CP&L has stated further that revision to NFP-NGGC-0003 to incorporate the burnup curve proposed as technical specification Figure 5.6.1 to reflect criticality screening requirements for fuel from all CP&L plants (Robinson, Brunswick, and Harris) to be stored in Harris pools C or D has begun. *Id.* However, this revision is not yet complete and will be put into production if CP&L's application to amend the Harris license to permit the licensee to place pools C and D in service is approved. *Id.*

Licensees have used administrative procedures to determine the acceptability for essentially all burnup-dependent storage pools since the early 1980's. *Id.*, ¶ 17. These procedures generally consist of verification that the licensee has selected a fuel assembly that has achieved the required amount of burn-up, based on plant operation records, and the licensee has stored it in the intended position in the spent fuel pool. *Id.* Section 4.2.1 of the American National Standards Institute (ANSI) standard ANSI/ANS-8.1-1983 states that nuclear criticality safety may be achieved by controlling one or more parameters of the system within subcritical limits and that control may be exercised administratively through procedures. *Id.* The NRC endorsed ANSI/ANS-8.1-1983 in revision 2 to Regulatory Guide (RG) 3.4. *Id.* In addition, 10 C.F.R. § 50.68 allows the use of administrative controls to prevent inadvertent criticality in fuel handling and storage. *Id.*, ¶ 18.

The industry uses administrative measures to prevent criticality in fuel storage and the NRC has accepted this practice since the early 1980's. *Id.* Further, since human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. *Id.*

Licensees have established their ability to predict core burnup behavior over hundreds of reactor years of operation. *Id.*, ¶ 19. They have also established their ability to predict isotopic inventories of reprocessed fuel by comparison of calculations from data available from several cores of the Yankee reactor. *Id.* To date, more than 50 plants have obtained NRC approval for the use of burnup credit for spent fuel storage. *Id.* To date, there have been no reported incidents of inadvertent criticality in U.S. spent fuel pools for any reason, including violation of administrative procedures. *Id.*, ¶ 18. In fact, there have been no instances where even the 5% subcriticality margin has not been maintained due to violations of administrative procedures. *Id.*

Draft Regulatory Guide 1.13 (Proposed Revision 2)(RG 1.13) recommends that the nuclear criticality safety analysis should demonstrate that criticality could not occur without at least two unlikely, independent, and concurrent failures. *Id.*, ¶ 22. This additional safety assurance is based on application of the "double contingency principle" as defined in ANSI/ANS-8.1-1983, which was endorsed by the NRC staff in a generic communication from Brian K. Grimes sent to all power reactor licensees on April 14, 1978. More recently, the Commission included similar criteria in 10 C.F.R. § 72.124(a), which requires two unlikely, independent, concurrent or sequential events to have occurred before a nuclear criticality accident is possible. *Id.*

The staff considers fuel misplacement in the Harris C and D storage racks to be an unlikely event for several reasons. *Id.*, ¶ 23. First, proposed technical specification 5.6.1.2 will control fuel storage limitations and selection procedure NFP-NGGC-0003, described above, will control fuel assembly selection. *Id.* Therefore, both technical specifications and plant procedures would have to be violated for a fuel assembly misplacement to occur. *Id.* In addition, fresh fuel assemblies have a bright, metallic color and are visually distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding. *Id.* Finally, the burnup limit curve proposed for the Harris technical specifications for acceptable storage in pools C and D is based on a minimum required burnup. *Id.* This is a bounding value that results in just meeting the 5% subcriticality margin in pools C and D. *Id.* In practice, unless an assembly is prematurely removed from the reactor, permanently discharged fuel assemblies would be expected to exceed these burnup requirements (have a lower reactivity). *Id.* Such fuel assemblies, therefore, should fall in the acceptable burnup domain of Figure 5.6.1, thereby minimizing the number of available fuel assemblies that could cause an increase in reactivity if misloaded. *Id.* Although there have been several reported fuel assembly misplacements at other plants in the past, the fact that these misplacements were reported and corrected indicates that administrative controls are effective in precluding permanent fuel misloadings. *Id.*

The placement of a fuel assembly in pools C or D that does not meet the technical specification burnup requirements and the continued failure to detect this misplacement is a highly unlikely event. *Id.*, ¶ 24. Multiple misplacements would be even more unlikely. *Id.* It is highly unlikely that a single failure in the administrative or the management process may lead to misplacement of multiple out-of-compliance assemblies. *Id.* Such a multiple misplacement, with or without boron dilution, leading to criticality, is highly improbable and well beyond the application of the double contingency principle. *Id.*

It is possible that loss of borated water could occur either by leakage or by overfill of the pool by unborated water. *Id.*, ¶ 25. However, attachment 1.2, sheet 10, of the Shearon Harris Chemistry and Radiochemistry Procedure CRC-001 specifies that the spent fuel pool boron concentration be maintained between 2000 and 2600 parts per million (ppm) and that the minimum boron concentration be confirmed by monthly surveillance measurements. *Id.* In addition, Harris technical specification 3.9.11 requires at least 23 feet of water above the top of the fuel rods. *Id.* Also, Final Safety Analysis Report (FSAR) Section 9.1.3 states that high and low level alarms are provided that would indicate water level changes and, therefore, potential dilution due to leakage or overfill by unborated water. *Id.* Visual indication of water level is also observed during each shift. *Id.* Therefore, the staff considers the loss of an appreciable amount of borated water from the pool to be highly improbable. *Id.*

The NRC staff requests a boron dilution analysis with respect to the spent fuel pools. *Id.*, ¶ 26. Standard Review Plan (SRP) 9.1.2 specifies that the reactivity of each spent fuel pool be at least 5% subcritical if moderated by unborated water. *Id.* This subcriticality margin is demonstrated in the criticality analysis for pools C and D of the proposed Harris amendment. *Id.* The analysis showed that k_{eff} in the proposed spent fuel pool C and D storage racks would be no greater than 0.95 if accidentally flooded with unborated water. This is an extremely conservative accident condition since the pool is about 25% or 30% subcritical under normal conditions with a minimum of 2000 ppm of boron and a complete boron dilution with loss of all soluble boron would be highly improbable for the reasons stated above. *Id.*

Holtec International performed the primary analysis of reactivity effects for the proposed use of Harris pools C and D. *Id.*, ¶ 27. In its analysis, Holtec analyzed reactivity effects of fuel storage in the Harris spent fuel racks using CASMO-3, which is a two-dimensional transport theory code. *Id.* Holtec also used CASMO-3 for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. *Id.* Holtec used the MCNP-4A Monte Carlo code to determine reactivity effects, to calculate the reactivity for fuel misloading outside the racks, and to determine the effect of having PWR and BWR racks adjacent to each other. *Id.* Holtec used MCNP-4A for independent verification calculations against CASMO-3. *Id.*

These codes (CASMO-3 and MCNP-4A) are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous criticality experiments. *Id.* (Benchmarking is the comparison of code predictions to known values for the purpose of validating the code. *Id.*) These experiments simulate the Harris spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. *Id.* In addition, these two independent methods of analysis showed very good agreement with each other. *Id.* Comparison of different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. *Id.* Moreover, these methods have been used and approved by the NRC staff in numerous other criticality analyses of spent fuel pools. *Id.*

In addition to the extremely conservative assumption of unborated water mentioned above, the Harris criticality analysis was performed with several other conservative assumptions that maximize the storage pool reactivity. *Id.*, ¶ 28. These include:

(a) Racks were fully loaded with the most reactive fuel authorized to be stored in the facility.

(b) Unborated water at the temperature yielding the highest reactivity over the expected range of water temperatures.

(c) Assumption of infinite array (no neutron leakage) of storage cells except for the assessment of peripheral effects and certain accidents.

(d) Neutron absorption in minor structural material is neglected.

(e) Uncertainties due to manufacturing tolerances were included to maximize the calculated k_{eff}.

(f) Calculational uncertainties and biases were incorporated to maximize the calculated k_{eff}.

Id.

Holtec Report HI-992283 presented the criticality evaluation of a fresh fuel misload in the Harris pools C and D. *Id.*, ¶ 29. This analysis determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain a 5% subcriticality margin in the event of a fuel assembly misloading event (*i.e.*, fresh fuel of 5% enrichment U-235 inadvertently placed in a location restricted to a burned assembly meeting the requirements of proposed technical specification Figure 5.6.1). *Id.* The results indicate that the minimum boron concentration of 2000 ppm required in the Harris spent fuel pools is more than adequate to offset the reactivity addition from a postulated fuel assembly misloading event. *Id.* Holtec also presented the results of an additional calculation in HI-992283, using the same NRC-acceptable methods, which showed that criticality would not be achieved for such a misloading event even for a concurrent accident condition of loss of all soluble boron. *Id.*

In addition, the NRC staff performed analyses to assess the possibility of unlikely misloading events leading to a criticality accident in the Harris C and D spent fuel pools. Ulses Affidavit, ¶ 6. For purposes of this analysis, the Staff assumed that soluble boron concentration was 2000 ppm, the pool water temperature was 4 degrees Celsius, and that the worst conceivable misloading would involve a Westinghouse 15 x 15 assembly enriched to

5 weight percent U-235 without burnable poisons, which is the highest allowed enrichment for commercial power reactor fuel and is bounding. *Id.* The Staff modeled the rack, fuel, and poison plate geometry using their nominal dimensions. *Id.*

The staff used the SCALE code system, which has been validated for these types of analyses, to perform the analysis. *Id.*, ¶ 7. The staff assumed that the storage racks were filled entirely with misloaded assemblies. *Id.* Such misloading could result only from multiple unlikely events, requiring multiple errors. *Id.* The maximum k_{inf} predicted for this configuration is 0.98. *Id.* Since this is less than 1.0, and the configuration studied represents the worst possible series of misloading events, the misloading of an entire rack of fresh fuel in spent fuel pools C or D will not lead to criticality. *Id.* This bounds the situation where there is only one misloading event because an entire misloaded rack has a much higher reactivity than a rack with only one assembly misloaded. *Id.*

3. <u>General Design Criterion 62 does not prohibit the use of credit for burnup to maintain subcriticality.</u>

General Design Criterion 62 provides:

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

10 C.F.R. Part 50, Appendix A. Nothing in Appendix A, GDC 62 prohibits the use of administrative methods to control the physical systems or processes referenced in that criterion. As set forth below, nothing in the history of GDC 62 prohibits the use of such

administrative controls, the Staff's consistent practice has been to allow licensees to rely, in part, upon such controls to satisfy that criterion, and the Commission has authorized the use of such controls relating to the prevention of criticality in spent fuel pools.

a. Rulemaking history

The Atomic Energy Commission (AEC)[7] added the General Design Criteria to Part 50 in 1971. The AEC went through an extensive process in drafting, redrafting and clarifying the GDC. As a result of these efforts, the AEC staff sent proposed criteria to the Commission in a paper dated June 16, 1967. "Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits," AEC-R 2/57 (Kopp Exhibit 2A). On July 11, 1967, the Commission formally published this revised version for comment. 32 Fed. Reg. 10,213. That version of the proposed GDC included Criterion 66, which provided: "*Prevention of Fuel Storage Criticality.* Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls." *Id.* at 10,216. Because the second sentence clearly contemplated the use of means such as geometrically safe configurations and procedural controls to prevent criticality, the statement in the first

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

statement that "criticality . . . shall be prevented by physical systems or processes" cannot be read to prohibit procedural or administrative controls.

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

> [w]e [do not] believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: "Such means as geometrically safe configurations shall be used to insure that criticality cannot occur."

Letter to H.L. Price, AEC, from W.B. Cottrell, NSIC, Enclosure (Specific Comments) at 11 (Kopp Exhibit 4).

Although there are no available staff documents discussing this comment, it is apparent that the Staff and the Commission did not agree with Oak Ridge that procedural controls should be prohibited, since the AEC did not adopt the suggested language. Rather, the AEC adopted the "preferably by use of geometrically safe configurations" language. *See* "Status Report on General Design Criteria, from Harold L. Price, Director of Regulation, NRC, to the Chairman and Commissioners, July 6, 1970 (Kopp Exhibit 4A); Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)(Kopp Exhibit 4B).

The staff again revised the criteria and the Commission adopted them as published in February of 1971.

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

In addition, GDC 62 applies to fuel handling systems, as well as fuel storage systems. While the fuel handling systems may move only one fuel assembly at a time, administrative controls must be used, for example, to prevent temporary storage of multiple assemblies in close proximity. Kopp Affidavit, ¶ 18. To adopt intervenor's reading of GDC 62, *i.e.*, that GDC does not allow the use of administrative controls to prevent criticality, would undermine the requirements to prevent criticality applicable to fuel handling, and should be rejected.[8]

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

(continued...)

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

Nearly every means to prevent criticality, and, in fact, just about every system or process in a plant has some administrative control associated with it, whether it is a surveillance, limiting condition of operation, testing, or some other administrative control. The intervenor objects to the use of credit for burnup in selecting the placement of spent fuel assemblies in the SFPs as an administrative control. However, because human action is necessary to move fuel between the reactor and fuel storage facilities, it is inescapable that administrative controls on fuel movement must be used to ensure that the physical measures for preventing criticality are properly employed. Kopp Affidavit, ¶ 18. Moreover, the burnup of the fuel is itself a physical process. *Id.*, ¶ 13. In addition, the Staff has been authorizing the use of credit for burnup in selecting fuel assemblies for locations in spent fuel

[8](...continued)

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

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

racks for eighteen years or more. *See id.*, ¶¶ 18, 19. There has never been a report of a criticality accident in any spent fuel pool. *Id.*, ¶ 18.

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

Moreover, at least one Atomic Safety and Licensing Board has accepted administrative controls to control the placement of fuel assemblies in spent fuel pools. *See Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 1), LBP-89-12, 29 NRC 441 (1989), *aff'd on other grounds*, ALAB-921, 30 NRC 177 (1989). The intervenor in that proceeding raised the following criticality[9] contention regarding misplacement of a fuel assembly:

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

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

> anywhere other than a design location," have not been demonstrated.

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

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

Id. at 456. Clearly, the Board in *St. Lucie* recognized that administrative controls are permissible to control criticality in a spent fuel pool.[10]

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

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

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

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

"Criticality Accident Requirements," 62 Fed. Reg. 63,911 (Dec. 3, 1997)(Proposed rule). The final rule included a similar statement. "Criticality Accident Requirements," 63 Fed. Reg 63,127 (Nov. 12, 1998)(Final rule). Responses to comments in the notice of issuance of the final rule explicitly demonstrate that the Commission was aware of licensee's use of administrative controls to prevent criticality in spent fuel pools. *Id.* at 63,128.

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

> The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity

> and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. The evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

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

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

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

As this regulation indicates, the Commission is aware of and has approved the use of administrative controls in conjunction with physical systems and processes as means of preventing criticality in spent fuel pools. Likewise, the use of administrative controls to prevent flooding with unborated water and optimum moderation of fresh fuel has also been approved. If the Commission has approved the use of administrative controls to control boron concentration to prevent criticality in spent fuel pools, then it is clear that the

Commission does not believe that the use of administrative controls in conjunction with physical controls violates GDC 62.

d. Other assertions by the intervenor

Dr. Thompson has asserted that dry cask storage is safer than spent fuel storage due to the absence of ongoing human actions and administrative controls attendant to dry cask storage. The staff submits that 1) the Commission has determined that both methods of storage are safe and 2) the assertion is clearly beyond the scope of the contention.[11] Therefore, there is no issue remaining as to Contention 2, Basis 1.

4. The use of credit for burnup is permissible. Criticality will not occur in the Harris spent fuel pools C & D without two independent failures, as specified in draft RG 1.13.

a. BCOC cannot expand the scope of the contention as admitted to this proceeding.

BCOC's contention was admitted with two bases. The second basis was defined by the Board as: "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,

[11] In Vermont Yankee, the intervenor, New England Coalition on Nuclear Pollution, which had proffered Dr. Thompson as an expert on the environmental contentions, agreed with the applicant therein that "the environmental impacts of dry cask storage and reracking are, apart from accident considerations, essentially benign and approximately equal." *Vermont Yankee Nuclear Power Corporation* (Vermont Yankee Nuclear Power Station), LBP-89-18, 29 NRC 539, 543 (1989). An Atomic Safety and Licensing Board has rejected similar assertions by Dr. Thompson with respect to a proposed contention regarding spent fuel pool criticality. *Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 2), LBP-92-28, 36 NRC 202, 218 (1992).

misplacement of a fuel assembly, could cause criticality."[12] *Shearon Harris*, LBP-99-25, 50 NRC at 36. The Board further refined the contention as asking 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?" *Id.* The scope of Contention 2, when read in context with basis two and the Board's Order, clearly limits the contention to consideration of a single misplaced fuel assembly.

Nevertheless, during discovery it became apparent that BCOC might seek to expand the scope of the contention by now contending that 1) a single failure could be a single failure that leads to multiple misplacements of fuel assemblies or 2) other failures, not specified in the contention could lead to criticality. The Staff submits that if BCOC asserts such theories, it would be an impermissible expansion of the scope of the contention, which

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[12] RG 1.13 is, of course, staff guidance and not a regulatory requirement. In 10 C.F.R. § 72.124, the Commission has applied the standard in RG 1.13 (with one change) to independent storage of spent nuclear fuel. Section 72.124(a) states:

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

The Staff does not know of any reason why storage of spent fuel in spent fuel pools at reactor sites should be handled any differently than for an independent spent fuel storage facility (ISFSI), as provided in section 72.124(a). The Staff applies the standard in that regulation to its evaluations of spent fuel pools.

should be denied by the Board. As stated by the Appeal Board in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29 (1989), a litigant is not free to:

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

Id. at 42. Furthermore, "intervenors are 'bound by the literal terms' of their own contentions." *Philadelphia Electric Co.* (Limerick Generating Station), ALAB-845, 24 NRC 220, 242 (1986) (citations omitted). The scope of a contention "necessarily hinges upon its terms coupled with its stated bases." *Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-98-7, 47 NRC 142, 181 (1998), citing *Public Service Co. of New Hampshire* (Seabrook Station, Units 1 & 2), ALAB-899, 28 NRC 93, 97 (1988). *See also Public Service Co. of New Hampshire* (Seabrook Station, Units 1 & 2), ALAB-947, 33 NRC 299, 372 (1991); *Limerick,* ALAB-845, 24 NRC at 242.

BCOC did not ask for reconsideration of the Board's ruling that admitted the contention, nor has it sought leave to expand the scope of the contention. BCOC, therefore, is limited to arguments based upon the contention and bases as admitted by the Board. Any attempt to expand the scope of the contention to assert other failures or that the Staff should consider multiple failures would be an impermissible expansion of the contention.

b. Criticality cannot occur in the Harris spent fuel pools C & D without two independent failures.

Basis 2 of Contention 2, as admitted, alleges that "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." LBP-99-25, 50 NRC at 36. As has been amply demonstrated in the applicant's submittal and analyses, and as reviewed by the Staff, the misplacement of one fuel assembly will not cause criticality (k_{eff} less than 0.95), assuming the spent fuel pool water contains 400 ppm of boron (which is less than the 2000 ppm of boron maintained in the pool water).[13] Kopp Exhibit 24, HI-992283 at 9; Kopp Affidavit, ¶ 29; Ulses Affidavit, ¶ 7. As has been further demonstrated in the licensee's analysis, two failures, that is, misplacement of one fuel assembly AND loss of all boron, will not cause criticality (k_{eff} less than 1.0). Kopp Exhibit 24, HI-992283 at 9; Kopp Affidavit, ¶ 29.

Although it is far beyond the scope of this contention and far more than the Staff normally requests by way of criticality analysis, the Staff performed an analysis of the effect of misplacing an entire rack of fresh fuel in the spent fuel pool. Such a misplacement would require multiple separate, independent failures. The results of that analysis are set forth in detail in the affidavit of Anthony Ulses, attached hereto. The analysis demonstrated that misplacement of an entire rack of fresh fuel assemblies would not cause criticality if 2000

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

ppm of soluble boron is present in the SFP water. Ulses Affidavit, ¶ 7. The loss of boron is a separate additional failure. Kopp Affidavit, ¶ 29. Therefore, the loss of boron need not be considered in any case.

As set forth above, the intervener is not capable of performing criticality analyses. In addition, both the Staff and CP&L have performed analyses demonstrating that a single fuel assembly misplacement, involving a fuel element of the wrong burn-up or enrichment, will not cause criticality in the fuel pool, and that even more than one such misplacement will not cause criticality. Rather, a misplacement coupled with some other error would be needed to cause such criticality. The Board should reject Basis 2 for Contention 2.

5. Conclusion with respect to Contention 2

Based on the foregoing, there are no genuine and substantial disputes of material fact as to any aspect of Contention 2 and there is no issue raised in Contention 2 that requires the introduction of evidence in an adjudicatory proceeding for resolution.

B. TECHNICAL CONTENTION 3 - QUALITY ASSURANCE

1. Summary of Facts, Data and Arguments

On December 23, 1998, CP&L submitted a request for a license amendment to place SFPs C and D into service. (Laufer aff., exhibit 1). The amendment request was accompanied by a request to use an Alternative Plan, under 10 C.F.R. § 50.55a(a)(3), to demonstrate an acceptable level of quality and safety for the component cooling system and SFP C and D cooling and cleanup system piping. (*Id.*, enclosure 8). As discussed in Section

III of this brief, the basis for the need for the Alternative Plan was that the construction of the cooling system was discontinued when Unit 2 was canceled and a formal turnover of the system was not performed. Some of the installation records for the piping and welds had been destroyed and the system cannot be N stamped. Therefore, the system will not satisfy ASME Section III code requirements. (*Id.*).

CP&L developed an Alternative Plan to demonstrate that the previously completed sections of the cooling system provide an acceptable level of quality and safety pursuant to 10 C.F.R. § 50.55a(a)(3)(i). The Staff evaluated the Alternative Plan against the criteria of that regulation.[14]

The Alternative Plan included a "Piping Pedigree Plan." (Laufer aff., exhibit 1, enclosure 8) The purpose of the Piping Pedigree Plan was to address the missing construction records by: 1) determining the boundaries of the piping within the plan; 2) reviewing and retrieving the existing construction era documentation, such as vendor data packages, work procedures, hydrostatic testing and concrete placement records, quality assurance records, and other relevant records; 3) examination and testing, where possible, of welds with missing records, using surface examination of accessible welds and visual examination of embedded welds; and 4) reviewing all data and documentation collected against the requirements of the ASME Code. (*Id.* at 3-4) The Licensee proposed to use an

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[14] 10 C.F.R. § 50.55a, requires that structures, systems, and components important to safety that must meet quality standards "commensurate with the importance of the safety function to be performed." *Codes and Standards for Nuclear Power Plants,* Final Rule, 49 Fed. Reg. 9711, 9712 (1984).

internal remote camera to perform the visual inspection of the embedded welds. (*Id.* at 3-11)

10 C.F.R. § 50.55a(a)(3)(i) permits alternatives to the ASME code to be used upon approval by the Director of the Office of Nuclear Reactor Regulation, if the alternative provides an acceptable level of quality and safety.[15] The staff used the 1974 Edition through 1976 Addenda of Section III of the ASME Code (Code), Section II as referenced therein (the Code applicable to the construction of HNP), and Appendix B of 10 CFR Part 50 in evaluating CP&L's plan. Section III requirements apply to construction of new components being placed in service. To determine the acceptability of the alternative plan, the Staff reviewed and evaluated information to determine if the alternative plan satisfied the Code or accomplished the same objectives as the Code requirements and to determine if applicable Appendix B requirements have been satisfied. This evaluation is necessarily done on a case by case basis, as it is fact specific, based upon the sections of the ASME Code and the details of the Alternative Plan. Such evaluations require knowledge of the ASME Code and the understanding of the purpose of the specific section to ensure that construction is adequate. The engineering judgments involved in this evaluation are based upon training, experience, a thorough understanding of the proposed plan, knowledge of the Code and its application, and an understanding of the purpose of the code sections. On this basis, the staff

[15] The Staff is not aware of anything in the history of 50.55a which further explains these criteria.

evaluated the suitability of the alternative plan to provide an acceptable level of quality and safety.[16]

The Staff's review and evaluation included examination of the application and any supplements thereto, all relevant supporting documents and data, review of QA/QC documents, welding documents and other relevant licensee documents, an on-site inspection, and review of video tapes of the inspections of the 15 embedded welds and the embedded pipes.

Based upon the review, the Staff concluded that the Alternative Plan provides an acceptable level of quality and safety. The records and other indicia reviewed by the Staff relating to the original construction of the piping and welds affords reasonable assurance that piping was constructed in accordance with the requirements of the ASME Code, 10 C.F.R. Part 50, Appendix B, and the licensee's approved QA program. The staff also concluded that there is reasonable assurance that the welds were completed with an acceptable level of quality and safety, no degradation of the welds or pipes occurred during the period when the system was in layup, and the pipes and welds are acceptable for use. These conclusions are supported in detail in the affidavits of Kenneth Heck, Donald Naujock and James Davis.

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[16] The methodology employed by the NRC staff in evaluating the alternative plan is the same methodology the staff would employ in evaluating other licensing actions, such as license amendment requests, in order to determine whether an amendment provides reasonable assurance that the health and safety of the public will not be endangered.

The Staff also concluded that if a failure of quality control led to a leak in the welds or piping, no hazard would occur. That conclusion is supported in the affidavit of Christopher Gratton.

Technical Contention 3 alleges as follows:

> CP&L's proposal to provide cooling of pools C & D by relying upon the use of previously completed portions of the Unit 2 Fuel Pool Cooling and Cleanup System and the Unit 2 Component Cooling Water System fails to satisfy the quality assurance criteria of 10 C.F.R. Part 50, Appendix B, specifically Criterion XIII (failure to show that piping and equipment have been stored and preserved in a manner that prevents damage or deterioration), Criterion XVI (failure to institute measures to correct any damage or deterioration), and Criterion XVII (failure to maintain necessary records to show that all quality assurance requirements are satisfied).
>
> Moreover, the Alternate Plan submitted by Applicant fails to satisfy the requirements of 10 C.F.R. § 50.55a for an exception to the quality assurance criteria because it does not describe any program for maintaining the idle piping in good condition over the intervening years between construction [and] implementation of the proposed license amendment, nor does it describe a program for identifying and remediating potential corrosion and fouling.
>
> The Alternative Plan submitted by Applicant is also deficient because 15 welds for which certain quality assurance records are missing are embedded in concrete and inspection of the welds to demonstrate weld quality cannot be adequately accomplished with a remote camera.
>
> Finally, the Alternative Plan submitted by Applicant is deficient because not all other welds embedded in concrete will be inspected by remote camera, and the weld quality cannot be demonstrated adequately by circumstantial evidence.

There are two distinct parts to the contention. First, there are the allegations relating to the failure to comply with Appendix B. Second, there are the allegations that the Alternative Plan is insufficient. In addition, there are three separate time frames covered by the contention: 1) the original construction of the embedded piping; 2) the conditions during the layup period; and 3) the present condition of the embedded welds and piping. Finally, there is the question of possible hazards if the criteria of 10 C.F.R. § 50.55a(a)(3)(i) - acceptable level of quality and safety - are utilized in analyzing the alternative plan. This brief will discuss each aspect in turn and demonstrate that there is no genuine and substantial dispute of material fact. In the alternative, any dispute which may remain can be resolved with sufficient accuracy without the introduction of evidence in an adjudicatory proceeding.

During discovery in this matter, it became clear that several issues originally encompassed by Contention 3 were no longer in contention. In his deposition of October 14, 1999, David Lochbaum, BCOC's sole expert witness regarding Contention 3, stated that the equipment which is intended for use in the cooling system for spent fuel pools C and D are not included within the ambit of Contention 3. Specifically, he stated that the heat exchangers are not part of the contention (Brief Exhibit 2, Lochbaum Tr. at 83). He acknowledged that the heat exchangers and pumps can be inspected to ensure that they meet the code and that they have not deteriorated. (*Id. at* 84). He further stated that the only issues in contention were the 15 embedded welds (and presumably the embedded piping).

He also stated that Contention 3 does not address the welds in the accessible piping. (*Id.*). According to Mr. Lochbaum, Contention 3 only concerns the embedded pipes and fifteen embedded welds. (*Id.* at 86-87). Based on the representations of BCOC's expert witness regarding the scope of their contention, the Staff will not be addressing the equipment or accessible welds and piping, other than such facts or data which may relate to the Staff's analysis of the embedded welds and piping.

a. Applicability of 10 C.F.R. Part 50, Appendix B to SFP C & D Cooling System

The first paragraph of Contention 3 asserts that CP&L's proposal to use the previously completed portions of the SPF cooling system fails to satisfy Criteria XIII, XVI and XVII of Appendix B to 10 C.F.R. Part 50. The licensee has admitted, and there is no dispute among the parties, that Criteria XIII and XVI were not complied with during the lay-up period.[17] The Alternative Plan was developed to address the missing records, and thus address Criterion XVII, which provides that "[s]ufficient records shall be maintained to furnish evidence of activities affecting quality. . . . Records shall be identifiable and retrievable.

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[17] Criterion XIII provides, as pertinent here, that "[m]easures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration."

Criterion XVI provides that "[m]easures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. . . ."

As set forth in the affidavit of Kenneth Heck, CP&L had an approved QA/QC program in effect during construction of the SFPs and the cooling system for SFPs C and D. (Heck aff. at ¶ 9). Thus, Appendix B was, in fact, applied during the construction stage. Because the cooling system for SPFs C and D was never completed and never placed in use, it was not licensed under the Unit 1 operating license. Unit 2 was canceled by CP&L on December 21, 1983. (Laufer aff. at ¶ 4). The Construction permit for Unit 2 expired on June 1, 1986. (*Id.*)[18]. At the time that Unit 1 was issued its operating license, in 1987, Unit 2 had been canceled and its CP had expired. Thus, the constructed portions of Unit 2 which had not been turned over, such as the cooling system for SFPs C&D were no longer covered by a construction permit or an operating license. Appendix B applies to "all activities during the design, construction, and operating phase of nuclear power plants which affect the safety-related functions of such structures, systems, and components." (*Quality Assurance Criteria for Nuclear Power Plants,* 34 Fed. Reg. 6599 (1969)). Since the piping and welds were no longer under construction, were not in operation, and had no safety related function during the lay-up period, the quality assurance provisions of Appendix B were no longer applicable to the SPF piping in question. Consequently, CP&L was no longer required to comply with Appendix B with regard to the unfinished, unlicensed cooling pipes and welds.

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[18] Units 3 and 4 were canceled by CP&L on December 16, 1981. The construction permits for Units 3 and 4 expired on June 1, 1990 and June 1, 1988, respectively. (Laufer aff. at ¶ 4).

The licensee will be required to comply with Appendix B and its approved QA program before putting the SFPs C and D and the cooling system into operation. Appendix B requires every applicant for a construction permit and every applicant for an operating license to include in their applications a description of the appropriate quality assurance program. It "establishes quality assurance requirements for the design, construction and operation of [safety related] structures, systems, and components." and is applicable to "all activities affecting the safety-related functions of those structures, systems, and components" including "designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying." (10 C.F.R. Part 50, Appendix B, Introduction). It stands to reason that once a permit or license expires, the requirements of Appendix B are no longer applicable. Therefore, CP&L does not have to demonstrate compliance with Appendix B during the lay-up period

As stated above, CP&L constructed Unit 1 pursuant to an approved QA program and has been operating Unit 1 under an approved QA program. There is no dispute that CP&L will be required to comply with its QA program and with Appendix B before placing the piping in service and all times thereafter.

CP&L's Quality Assurance and Quality Control programs were approved by the Staff and that approval was affirmed by the Licensing Board at the Construction Permit stage. *See Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4), LBP-78-4, 7 NRC 92, 107-09 (1978). The Licensing Board held that: "The QA programs describe adequate QA procedures, requirements, and controls demonstrating that quality-related activities will be conducted in accordance with the requirements of Appendix B of 10 CFR Part 50." *Id.* at 108.

BCOC's witness, David Lochbaum, cited four notices of violations relating to construction QA in his affidavit in support of the request for hearing. They were cited in support of his assertion that the licensee had QA problems with proper storage of equipment. The Staff notes that the violations cited were level V and VI violations, they appear to be isolated incidents over an extended period, and they were appropriately dealt with by the licensee. (Heck aff. at ¶ 38). These isolated incidents offer absolutely no support for BCOC's thesis that CP&L's QA was inadequate during the construction of the embedded pipes or that there is no reasonable assurance that the welds and piping were adequately constructed. As the Appeal Board has said:

> In any project even remotely approaching in magnitude and complexity the erection of a nuclear power plant, there inevitably will be some construction defects tied to quality assurance lapses. It would therefore be totally unreasonable to hinge the grant of an NRC operating license upon a demonstration of error-free construction. Nor is such a result mandated by either the Atomic Energy Act of 1954, as amended, or the Commission's implementing regulations. What they require is simply a finding of respnable assurance that, as built, the facility can and will be operated without endangering the public health and safety.

Union Electric Co. (Callaway Plant, Unit 1), ALAB-740, 18 NRC 343, 346 (1983). *See also Georgia Power Co.* (Vogtle Electric Generating Plant, Units 1 and 2), ALAB-872, 26 NRC 127, 141 (1987); *Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 728-29 (1985); *Pacific Gas & Electric Co.,* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-756, 18 NRC 1340, 1345 (1983). In *Diablo Canyon,* the Appeal Board said:

> It is, of course, possible that a review of the work of the remaining contractors might lead to the discovery of serious construction or construction quality assurance flaws. But the theoretical possibility of such discoveries is insufficient. To demonstrate the need for additional construction quality review, the movants must either establish construction errors that endanger safe plant operation or show a pervasive failure of the quality assurance programs sufficient to raise legitimate doubts as to the adequacy of a plant's construction.

Id. at 1350 . In *Vogtle*, the Appeal Board rejected the intervenor's argument that the deficiencies cited were "evidence of a pattern that suggests a high likelihood that some structure, system or component will eventually fail." *Vogtle*, ALAB-872, 26 NRC at 141.

In the *Callaway* case, the Intervenors complained about installed embedded plates (embeds) which were not retested after possible discrepancies were found in the testing procedures for the welds on the embeds. All not yet installed welds were reinspected. *Callaway*, ALAB-740, 18 NRC at 352-53. The inspection revealed that the welds were undersized. *Id.* The load design capacity for the installed plates was recalculated. *Id.* Installed plates were visually inspected. *Id.* at 354. The appeal board found that visual inspection combined with expert evaluations of the code requirements and the maximum loads on the embeds, and the evaluations of the uninstalled embeds, was sufficient to support the licensing board's finding that the embeds were safe. *Id.* at 354-56. In the instant case, the accessible welds have been fully examined. Licensee experts and staff experts have evaluated the code requirements and the available records and evidence relating to the welds, and a detailed visual inspection of the embedded welds was performed by the licensee and evaluated by the NRC staff. All these factors are sufficient to demonstrate the safety of the embedded welds. BCOC has not shown that there are any legitimate, substantial issues as to the quality of the original construction of the embedded piping and welds. Therefore, there are no genuine and substantial material facts in issue and thus no reason to hold an evidentiary hearing regarding Contention 3.

b. The Alternative Plan is sufficient to assure an acceptable level of quality and safety.

The alternative plan provides an alternative to the ASME code to demonstrate that the welds and piping for which the documents are not available are acceptable for service. The plan also addresses the method by which CP&L will demonstrate that there is reasonable assurance that the welds and piping are acceptable for service in their present condition. The Staff has reviewed both the Alternative Plan *and* the results of the licensee's examination of the welds and piping.

There are two areas which are addressed in evaluating the Alternative Plan and the suitability of the piping and welds for service in Unit 1: the missing records regarding the original construction of the piping and welds and the present condition of the piping and welds. The details of the Staff's review of the missing records and the Alternative Plan are discussed in the affidavits of Kenneth C. Heck and Donald Naujock. The present condition of the piping and welds is discussed in the affidavits of Donald Naujock and James Davis.

(1) Original Construction of the Embedded Piping and Welds.

The Staff's review and the results of its evaluation of the quality assurance program in place at the time of original construction and the available quality assurance documentation is contained in the affidavits of Kenneth C. Heck and Donald G. Naujock. Mr. Heck reviewed the weld records and the acceptability of the piping for use in Unit 1. Based upon his review, he concluded that there was reasonable assurance that the embedded welds were completed in accordance with the applicable regulatory requirements and provide an acceptable level of quality and safety. Mr. Naujock reviewed the adequacy of the alternative plan for the welds with missing documentation. He evaluated the quality assurance (QA) procedural and program controls, the material selection for the welds, the set-up of the weld joints, the importance of welder symbol documentation, the inspection of the piping welds, and the weld repair documentation. Based on his review and evaluation, he concluded that CP&L's alternative plan for addressing the SFPs C & D piping and welds with missing documentation demonstrates an acceptable level of quality and safety.

Both Mr. Heck and Mr. Naujock participated in the Staff's November 15 -19, 1999, on-site inspection to assess the implementation of the construction quality assurance program in the construction of C and D spent fuel pools.

Mr. Heck reviewed the licensee's plan against the guidance of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 17, "Quality Assurance," and Regulatory Guide (RG) 1.26.

The licensee submitted the quality assurance (QA) program for control of ASME code-related activities that was in effect during construction, and additional QA provisions for completion of the cooling system. The construction QA program had been approved by the Staff during the construction phase. (Heck aff., Exhibit 19).

The staff reviewed the Staff approved CP&L QA program in effect during construction and concluded that Unit 1 (SFPs A and B) and Unit 2 (SFPs C and D) were constructed under a common quality assurance program, using a single source of oversight and construction personnel. (Heck aff. at ¶¶ 7, 19) The both shared a common design basis which was concurrently reviewed and approved by the staff in NUREG-1038. (Heck aff. at ¶ 19) These facts provide reasonable assurance that the quality of construction for Unit 2 was similar to the quality of construction for Unit 1.

The Staff reviewed the existing records required under the ASME code for the completion of the piping for the cooling system for Unit 1 in order to determine the original conditions under which the piping and welds were constructed. (Heck aff. at ¶¶ 21, 23)

Those records were found to be complete. The QA program and procedures relating to the welding data reports were found to be clear and sufficiently detailed to demonstrate compliance with the Code reporting requirements. (*Id.* at ¶¶ 24-32) The ASME Code requires that Owner's Data Report (N-3) be prepared for completed units. The N-3 data package for Unit 1 was examined and was found to contain properly completed N-5 forms for piping used in Unit 1. The Code required completion of form N-5, which verifies the field installation of welds. (*Id.* at ¶ 23) All the packages examined were signed by the licensee and the Authorized Nuclear Inspector (ANI)[19] and filed with and received by the National Board of Boiler and Pressure Vessel Inspectors. (*Id.*) The records reviewed demonstrated that the welding of the cooling piping for Units 1 and 2 was done during the same time period, by a common pool of welders in accordance with a common system of procedural requirements. (*Id.*) Therefore, the staff concluded that the acceptance of the N-stamp registration for the welds for the Unit 1 pools provides substantiating evidence that the welding of similar piping for Unit 2 was of comparable quality.

The Staff reviewed N-5 data packages for Unit 2 equipment, (other than the piping and welds), such as heat exchangers and pumps. The records reviewed were found to be complete.

The Staff reviewed the available construction records for, *inter alia,* the 15 embedded welds in contention. The installation of piping and welds must be done in sequence, with

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[19] An independent reviewer employed by the state, municipality or insurance company.

each step being completed before the next step can begin. (*Id.* at ¶¶ 29, 37) The records of the hydrostatic tests for 13 of the 15 embedded welds were available. Before the hydrostatic tests can be performed, all prior installation steps must have been performed in compliance with the ASME Code and other quality requirements, and approved by the QC/QA staff and the ANI. Based on the review of the hydrostatic test records, the Staff concluded that because the hydrostatic testing records indicated that the QC/QA staff and the ANI had signed off that all steps had been completed, the hydrostatic test records provide reasonable assurance that the welds were completed in compliance with applicable quality standards before hydrostatic testing. (*Id.* at ¶ 37).

The Staff reviewed the QA audit program, which verified the effectiveness of the QA program. The records were found to be complete and adequate and provided substantiating evidence of an effective ASME program and adequate corporate oversight. (*Id.* at ¶ 24) The Staff review of the vendor audit program found it to be similarly complete and adequate. (*Id.* at ¶ 25)

The Staff reviewed the QA manual and the QA/QC implementing procedures and found that they provided adequate control of the weld process to provide reasonable assurance that the field welds met the code requirements, hydrostatic testing to provide reasonable assurance that the weld records had been satisfactorily completed and accepted before commencement of hydrostatic testing, and concrete placement to provide reasonable assurance that the welds were completed, documented and found to be in compliance with the applicable code and quality requirements before being embedded in concrete. (*Id.* at ¶¶ 26-27).

The Staff also examined samples of deficiency and disposition reports (DDRs), nonconformance reports and reports of QA/QC monitoring and surveillance of field activities. The Staff concluded that the documentation provides substantiating evidence that the quality assurance program during the period of welding had been effectively implemented. (*Id.* at ¶ 34).

The Staff reviewed the welding control procedures to verify that weld activities and processes were controlled in accordance with applicable code requirements. Based on examination of these procedures, the staff concluded that at the time of original construction of the existing fuel pool cooling system, a comprehensive welding program was in place to control and document pipe welding in accordance with Section III of the ASME Boiler and Pressure Vessel Code. (*Id.* at ¶ 28).

The Staff reviewed records of the recent reinspection of all accessible field welds for the Unit 2 fuel pool cooling system and associated component cooling water system pipe and pipe attachments. Reinspection included visual and liquid penetrant examination, recording of welder identification, and verification of welder qualification. The information was used to create new weld data reports for the accessible welds for which documentation is missing. (*Id.* at¶ 36). In addition, the Staff reviewed, weld records for Unit 1 welds. These welds were made using the same welding QC program, during the same construction period, as the

Unit 2 welds. The original construction records were retrievable, legible, and complete. The Staff concluded that the records provided evidence that an effective quality program had been implemented during original construction. (*Id.* at ¶ 23).

The Staff reviewed NRC inspection reports for the construction period from 1978 through 1983. Several deficiencies dealing with the general subject of welding were identified in these reports. Most of these deficiencies were relatively minor (Severity Level V and VI) and would not be cited under the current inspection program and would be resolved through the licensee's corrective action program. All deficiencies were typical of what one would expect for oversight of a large construction project and are not indicative of any programmatic weakness in the licensee's weld program. (*Id.* at ¶ 39).

Mr. Naujock reviewed and evaluated the Alternative Plan using the 1974 Edition through 1976 Addenda of Section III of the ASME Code (Code) and Section II, and 10 C.F.R. Part 50, Appendix B. He evaluated the plan against specific code requirements in order to determine whether the alternative plan satisfied the Code or accomplished the same objectives as the Code requirements and to determine if applicable Appendix B requirements have been satisfied. In making these determinations, he relied on his experience in applying Code requirements and his knowledge of the Code and the purpose of specific Code requirements. In addition, he relied on his knowledge of the Alternative Plan and knowledge of the SFP piping and welds. He applied his engineering judgment and knowledge to all the factors. On that basis, he evaluated the suitability of the alternative plan to provide an acceptable level of quality and safety. A determination of the acceptability of an alternative to Code requirements is conducted on a case-by-case basis.[20] Each alternative must be evaluated based upon its unique facts and requires application of the Code requirements and objectives to the specific facts of each situation. (Naujock aff. at ¶ 6).

The Staff reviewed the data and analyses for accessible Unit 2 welds and for Unit 1 welds, relating to specific requirements of the Code, including the chemical composition of the weld metal or filler metal, ferrite percentage in the weld filler, tack welds, alignment of piping sections, welder identification symbols and weld repair documentation and found them all to be acceptable pursuant to the ASME Code. (*Id.* at ¶¶ 10-14, 16). Based on the data reviewed, the Staff concluded that the alternative plan provided an acceptable level of quality and safety with respect to those Code requirements. (*Id.*)

The Staff concluded that the field welds were completed to an acceptable level of quality and safety and that there is substantial documentation that supports the conclusion that the subject welds were completed with an acceptable level of quality and safety. The Staff also concluded that the alternative plan for the piping and welds with respect to the 15 embedded welds with missing documentation either satisfied the Code requirements or provided an acceptable substitute, thereby demonstrating an acceptable level of quality and

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[20] The methodology employed by the NRC staff in evaluating the alternative plan is the same methodology the staff would employ in evaluating other licensing actions, such as license amendment requests, in order to determine whether an alternative provides reasonable assurance of protection of health and safety and whether the Code objectives are met.

safety. (*Id. at 17).* The Unit 1 fuel pool has supported Unit 1 operation since the beginning of commercial operation in 1987 and has operated without significant problems for more than twelve years.

(2) Present Condition of the embedded pipes and welds.

Before being permitted to place the cooling system in service, CP&L must demonstrate that the piping and welds are safe for their intended purpose. The NRC Staff evaluated CP&L's commissioning plan for the piping and the alternative plan and evaluated the analyses and examinations of the embedded pipes and welds, their environment, and their present condition. The examination and evaluation was performed by Donald Naujock and James Davis.

In order to substantiate the validity of the surface inspection during construction and to recreate missing surface inspection welds, CP&L reinspected the accessible welds to Code requirements and no reportable flaws were observed. Since an exterior surface inspection of the embedded welds was impractical and since the Code does not specify whether the inner or outer surface of the welds are to be inspected, CP&L proposed an alternative to the code required magnetic particle, liquid penetrant or radiographic inspection. CP&L visually examined the interior surface of the embedded welds using a remote camera. Enhanced visual inspection using a high resolution video camera in place of surface examination has been previously approved by the Staff for reactor vessel internals. The equipment and qualifications of the operators was reviewed and accepted by the Authorized Nuclear

Inspectors. They also observed a demonstration of the equipment. The camera is able to detect a 1 mil diameter wire and was able to detect flaws in the demonstration. A description of the remote camera procedure and qualification is found in Dr. Davis' affidavit at ¶ 8 -9, and Mr. Naujock's affidavit at ¶ 15. The Staff concluded that the remote visual inspection was capable of detecting flaws consistent with Code requirements.

Dr. Davis reviewed the video tapes of the remote examinations of the embedded welds and observed that ten welds had no evidence of biofouling (microbiologically influenced corrosion - MIC), or degradation of the circumferential weld, the heat affected zone or the base metal. These welds required no corrective action. Five welds required further assessment using the HNP Appendix B Corrective Action Program. The welds were further analyzed by an independent reviewer, Structural Integrity Associates (SIA). (Davis aff. at ¶¶ 10-12). Based on review and analysis of the video tapes, SIA's data and analysis, the available hydrostatic testing documentation, and the results of CP&L's corrective actions, the Staff concluded that the piping and welds: are conservatively designed and several times thicker than required by the Code; are generally in good condition, with some minor defects, but no major defects; and have leak tight integrity. (*Id.* at ¶¶ 12-14) In addition, the Staff concluded that there were no viable mechanisms for a longitudinal crack, suchh as, intergranular stress corrosion cracking (IGSCC) or transgranular stress corrosion cracking (TGSCC). It was also concluded that there was no viable mechanism, such as IGSCC, TGSCC, or localized corrosion, for degradation of the piping. The only viable mechanism for corrosion is MIC. Water samples and visual inspection of the piping and sampling of a deposit on a weld produced no evidence of MIC. In addition, after deposits were removed, the weld was reexamined an no damage was observed. No leaks consistent with MIC were observed in the accessible piping. (Davis aff. at ¶¶ 15-19). The Staff concluded that the remote, enhanced visual inspection can be used to detect flaws and that a sufficient basis exists to state with reasonable assurance that the welds were completed with an acceptable level of quality and safety and no degradation of the welds and pipes occurred during the layup (*Id.* at 21).

The Staff therefore, concludes that the welds and piping are acceptable for service, and that the Alternative Plan provides and acceptable level of quality and safety. Therefore, there are no genuine or substantial material facts in issue and there is no issue remaining which requires the introduction of evidence in an adjudicatory proceeding for resolution.

2. BCOC's Designated Expert Witness, David Lochbaum, Should Be Disqualified
As an Expert Witness and his Testimony/Declaration Limited or Stricken

BCOC proffered David Lochbaum as its expert witness for Technical Contention 3, inadequate quality assurance. The staff submits that Mr. Lochbaum has not demonstrated that he has the requisite expertise to address the issues related to Contention 3.

As discussed in Section V.A.1. *supra.,* persons who seek to present expert testimony must be qualified to do so. Mr. Lochbaum is, by his own admission, not qualified to give expert testimony in the following areas relevant to Contention 3: material science, corrosion of materials, stress analysis, failure analysis, causes of degradation of stainless steels, or probability and statistics as applied to engineering design. (Exhibit 2, Lochbaum Dep. at 41-43). In addition, he does not have experience as a construction engineer or welding engineer or in welding or non-destructive examination (NDE). He has limited experience in Quality Assurance and Quality Control. (Exhibit 2, Lochbaum Dep. at 38-41).

BCOC has not produced any information which would indicate that Mr. Lochbaum possesses the necessary expertise in any of the technical areas implicated in Contention 3. He has no training or experience in welding, corrosion, materials science, or related topics, he has limited experience in QA and QC. Since this Contention deals with the adequacy of QA and QC and the adequacy and condition of welds and pipes, it is difficult to determine how Mr. Lochbaum will be of assistance to the Board in deciding Contention 3. Therefore, any of his testimony which is based on opinion in these areas should be stricken.

<u>Failure of Quality Control Will Not Lead to a Hazard.</u>

In admitting Contention 3, and in discussing the criteria in 10 C.F.R. § 50.55a(a)(3)(i), the Board stated: "[a]nd, of course, if CP&L's plea is that the proposed alternatives provide an acceptable level of safety, we will need to confront directly the question of whether a failure of quality control could lead to a hazard." *Harris*, LBP-99-25, 50 NRC at 37. The Staff evaluated the proposed alternatives against the criteria of 10 C.F.R. § 50.55a(a)(3)(i). The following discussion addresses this issue. The Staff has concluded that, based on the design of the SFPs and the cooling system, the likelihood that a failure of quality control (QC) will lead to a hazard is minimal.

The question of possible hazards is discussed in the affidavit of Christopher Gratton, a reactor systems engineer assigned to review the CP&L amendment request. Mr. Gratton reviewed the design and considered the operation of the spent fuel storage system at Harris and concluded that in the event that a failure of quality control results in the failure of a passive component, it is unlikely that a hazard that affects public health and safety will result. (Gratton aff. at ¶¶ 4, 17). If a failure of an embedded weld occurs where the leakage cannot flow out of the pool's concrete structure, this failure will have minimal effect on the operation of the fuel pool cooling and cleanup system, the coolant inventory in the spent fuel pool, or the safety of the stored fuel. (*Id.* at ¶ 8). If a failure of an embedded weld occurs where the leakage is able to flow out of the pool's concrete structure but whose leakage is within the capacity of the coolant make up systems, once detected by the plant operators via the multiple alarms and other indicia of loss of coolant, the failure would be mitigated by plant operators who would maintain the spent fuel pool at its normal operating level and the repair the damaged piping. (*Id.* at ¶¶ 9, 10). This type of leak may have a temporary effect on the operability of the fuel pool cooling system, but would not affect the safety of the stored fuel. If a leak greater than the capacity of the coolant makeup systems developed in an embedded portion of the fuel pool cooling and cleanup system, the spent fuel pool would drain to level equal to the fuel pool cooling and cleanup system piping penetration (approximately 18 feet above the stored fuel) causing a loss of all forced cooling to the affected spent fuel pool. The pool would gradually heat up, and if repairs to the damaged pipe could not be made in sufficient time, the pool would begin to boil. However, due to the low decay heat rate of the stored fuel, the rate of boiling would be low and within the capacity of the available coolant makeup systems to maintain the coolant inventory.[21] (*Id.* at ¶¶ 16, 17). Maintaining the coolant inventory ensures the fuel cladding will not overheat, become damaged and create a possible hazard that affects public health and safety. Therefore, in each postulated scenario where a degraded weld fails resulting in a leak from the spent fuel pool cooling system, the stored fuel remains covered and cooled with only a minimal impact on public health and safety. (*Id.* at ¶ 17).

As demonstrated in Mr. Gratton's affidavit, multiple, successive, unlikely failures would have to occur before a hazard to public health and safety could result from a leaking embedded weld. These assumptions are far beyond any credible event at this plant. Although the loss of cooling could eventually lead to boiling, there are sufficient redundancies, alarms, and other safety features to make such an occurrence improbable. The Appeal Board has pointed out that such accidents (complete loss of pool water caused by loss of cooling water circulation capability) have a very low likelihood and do not contribute

[21] Boiling point would not be reached for approximately 300 hours after loss of cooling and the estimated rate of boil off is only a few gallons per minute. (Gratton aff. at ¶ 16 and 17). Lesser time frames have been found to be acceptable. *See Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 1), LBP-88-27, 28 NRC 455, (1988). In that case, the SFP was calculated to reach the boiling point 5 to 13 hours after a loss of cooling capability and 46 hours thereafter to boildown enough to uncover the fuel. *Id.* at 469.

significantly to risk. *See Vermont Yankee Nuclear Power Corporation* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29, 45-46 (1989). Any leak in an embedded weld would likely cause an extremely slow leak. Any loss of water from such a leak could be made up by the available sources of make-up water.

Based upon the foregoing, the Staff concludes that a failure of quality control affecting the cooling system embedded welds and piping will not cause a hazard. Therefore, there are no genuine and sustantial disputes of material fact as to possible hazards and there are no issues which require the introduction of evidence at an evidentiary hearing.

CONCLUSION AS TO TECHNICAL CONTENTION 3

The Staff concludes that the Licensee's Alternative Plan provides an acceptable level of quality and safety. The records and other indicia reviewed by the Staff relating to the original construction of the piping and welds affords reasonable assurance that piping was constructed in accordance with the requirements of the ASME Code, 10 C.F.R. Part 50, Appendix B, and the licensee's approved QA program. The licensee was not required to demonstrate compliance with Appendix B during the layup period when the cooling system was not covered by a license. The staff also concludes that there is reasonable assurance that the welds were completed with an acceptable level of quality and safety, no degradation of the welds or pipes occurred during the period when the system was in layup, and that the pipes and welds are acceptable for use. Staff concludes that a failure of quality control would not cause a hazard.

The documentation and data supporting these conclusions are attached hereto as affidavit exhibits. The Licensee has demonstrated that it meets the criteria of 10 C.F.R. § 50.55a(a)(3)(i) for authorization of an alternative plan for complying with ASME Code requirements. The Staff has concluded that the welds and piping did not degrade during the layup period. Finally, the Staff has concluded that the welds and piping are acceptable for use. The Staff submits that there are no further facts which need to be developed or which require the introduction of evidence in an adjudicatory proceeding for resolution. There are no genuine and substantial disputes of material fact as to any aspect of Contention 3.

CONCLUSION

Based upon the foregoing, the Staff submits that there are no genuine and substantial disputes of material fact as to any aspect of Contentions 2 or 3 and there is no issue raised in the contentions which require the introduction of evidence in an adjudicatory proceeding for resolution.

Respectfully submitted,

Susan L. Uttal
Counsel for NRC staff

Robert M. Weisman

Robert Weisman
Counsel for NRC staff

Dated at Rockville, Maryland
this 4th day of January 2000.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON TECHNICAL CONTENTIONS 2 AND 3," supporting affidavits and exhibits, and "NOTICE OF APPEARANCE" in above-captioned proceeding have been served on the following by hand delivery, and by deposit in the U.S. Postal Service as indicated by an asterisk, this 5th day of January, 2000:

G.Paul Bollwerk, III, Chairman
Administrative Judge
Atomic Safety and Licensing Board
Mail Stop: T 3F-23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Peter S. Lam
Administrative Judge
Atomic Safety and Licensing Board
Mail Stop: T 3F-23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Office of the Commission Appellate Adjudication
Mail Stop: O 16C-1
U.S. Nuclear Regulatory Commission
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Frederick J. Shon
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Susan L. Uttal
Counsel for NRC Staff

Brief Exhibit 1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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
GORDON THOMPSON, PH.D.

At Raleigh, North Carolina
October 21, 1999
9:40 AM to 4:14 PM
Reported by: Melody L. Rife, RPR

COPY

APPEARANCES

For the Applicant, Carolina Power & Light Company:

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For the Intervenor, Board of County, Commissioners, Orange County, North Carolina:

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Washington, D.C. 20036
(202) 328-3500

Also Present: Messieurs Michael Devoe, John Caves, Richard Laufer, and Laurence Copp

S T I P U L A T I O N S

It is hereby stipulated and agreed between the parties to this action, through their respective counsel of record:

1. The deposition of Gordon Thompson, Ph.D., may be taken on October 21, 1999, beginning at 9:30 AM, at the offices of Carolina Power & Light Company, Fayetteville Street Mall, Central Plaza Building, 13th Floor, Raleigh, North Carolina, before Melody L. Rife, Registered Professional Reporter and Notary Public.

2. Any objections of any party hereto as to notice of the taking of said deposition or as to the time or place thereof, or as to the competency of the person before whom the same shall be taken are deemed to have been met.

3. Said deposition shall be taken for the purpose of discovery or for use as evidence in the above-entitled action or for both purposes.

4. Objections to questions and motions to strike answers need not be made during the taking of this deposition, but may be made for the first time during the progress of the trial of this case, or at any pretrial hearing held before any judge for the purpose of ruling thereon, or at any other hearing of said case at which said deposition might be used, except that an objection as to the form of a question must be made at the time such question is asked or objection is waived as to the form of the question.

5. The original of this deposition will be mailed to the appropriate party. Notice of filing is hereby waived.

6. Deponent reserves the right to read and sign the deposition.

* * * * *

Thereupon, the following proceedings were had:

* * * * *

(Thereupon, a discussion was held off the record)

DR. HOLLAWAY: I also ask that you transcribe everything during the deposition, except during breaks and when we go off the record, when nothing should be transcribed. And please interrupt, if it's necessary, to clear up any doubt about a question or answer.

THE COURT REPORTER: Thank you.

DR. HOLLAWAY: I'd like you to mark exhibits prior to commencing examination, so we have that clear.

(Thereupon, a discussion was held off the record)

* * * * *

Thereupon,

GORDON THOMPSON, PH.D.

having first been duly sworn, was examined and testified as follows:

things?
A. At present, yes, I'm --
Q. Okay.
A. -- looking for that sort of document.
Q. Have you identified anything yet?
A. I intend to get the American Nuclear Society standards on criticality, and I don't have those in my files as of yet.
Q. Okay. Anything else?
A. Nothing that I'm actively looking for at present.
Q. Okay.
When were you retained by BCOC with respect to this proceeding?
A. My recollection is January, but it may not be exact. I'd have to consult my files.
Q. Of what year?
A. This year, 1999.
Q. Okay. And what is your role with respect to this proceeding, as you understand it?
A. To provide technical and safety advice to the County pursuant to its intervention in the license application for the fuel

expansion, working with Attorney Curran, who works for the County, also.
Q. Do you understand that in this proceeding, Counsel has filed a pleading stating that you will be an expert with respect to Contention 2 only?
A. Yes, I understand it.
Q. Okay.
Are you being compensated to be here today?
A. Yes.
Q. And who is paying you?
A. Directly, my employer, the Institute for Resource and Security Studies. In turn, they are compensated by Orange County.
Q. How much is the Institute for Resource and Security Studies being compensated for your work here today?
A. My time is billed at an hourly rate, and whatever that adds up to.
Q. What is that hourly rate?
A. Hundred and twenty-five dollars per hour.
Q. How many hours do you expect to spend

A. Right.
Q. Do you agree with the findings of this book?
A. I find it a generally useful book that I found to contain generally accurate information. I would not necessarily support all of the findings and recommendations.
Q. Any findings or recommendations that you know of that you don't agree with in Mr. Lochbaum's book?
A. I don't recall any at present.
DR. HOLLAWAY: I'll ask the court reporter to mark as Exhibit 2 the curriculum vitae of Gordon R. Thompson dated July 1999.
(Thereupon, Thompson Exhibit No. 2 was marked for identification)
Q. Dr. Thompson, have you seen this document before?
A. I wrote it.
Q. So you authored this.
A. Yes.

Q. Are the statements in here truthful?
A. Yes.
Q. This states that you have a Ph.D. in applied mathematics?
A. Correct.
Q. What does that relate to?
A. The work was in the -- the theory of high-temperature plasmas. So it could be considered theoretical physics, but it happened to be done through the math faculty.
Q. Can you tell me what courses you have taken in fission reactor engineer?
A. None.
Q. Can you tell me what courses you've taken in fission reactor criticality control?
A. None.
Q. Okay. What training have you had in fission reactor criticality analysis?
A. None.
Q. Are you an expert in fission reactor criticality analysis?
A. For the purpose of this proceeding, yes.

Q. On what basis do you state that?
A. My contribution to the -- to this proceeding relies on my basic expertise in scientific principles and analytic principles and my general experience with engineering in general and nuclear plant engineering in specifics.
Q. So when you assert that you're an expert in fission reactor criticality analysis, that would be in the general scientific principles attendant to criticality?
A. The brief that -- to which I will -- that -- my contribution to Orange County's brief will rely upon expertise that I possess.
Q. Could you answer my question?
THE WITNESS: Could you read it back?
(Thereupon, the question beginning on page 21, line 10, was read by the court reporter)
A. Yes, and on the application of those principles to the contention.
Q. Okay.

Tell me what criticality analysis codes you have run yourself.

A. I have not run any, as such.

Q. Okay. Can you tell me what training you've had in running criticality analysis codes?

A. None.

Q. Okay. What codes are used to perform fission reactor criticality analysis?

A. Codes that are identified in the CP&L application and in the subsequent correspondence, response for the request for additional information.

I don't remember the names of those codes. And I should say as a point of clarification that I don't expect to run or seek to have run any of those codes in connection with this proceeding.

Q. Okay, so you have not run any criticality analyses yourself for this proceeding?

A. Correct, and do not anticipate doing so or having this done.

Q. Okay. Are you competent to evaluate the results of a criticality analysis?

A. Yes.

Q. If you've never been trained in running the codes, have not run the codes yourself, how can you evaluate whether the analysis itself is correct?

A. In evaluating an analysis, there are two primary aspects to the evaluation. One is to -- given the assumptions on the line analysis, to assess the analysis that was performed pursuant to those assumptions. The other aspect is to examine the assumptions and assess whether those assumptions are sufficient to address the issues that might be of concern in connection with criticality.

I -- in the course of this proceeding, I will expect to confine my assessment primarily and perhaps totally to the assessment of assumptions and their adequacy.

Q. So you've identified two aspects here. The first one is sufficiency of the assumptions --

A. Right.

Q. -- second is given those assumptions, the analysis itself.

A. Correct.

Q. You believe that you're competent to address the sufficiency of the assumptions; is that correct?

A. Yes.

Q. Do you have the expertise to address the second part, whether -- given those assumptions are valid, that the analysis done after it is in fact correct and valid?

A. Not without doing a lot of studying. As of this moment, no, I am not competent to do that.

Q. Okay. Do you anticipate doing that?

A. Not over the time frame of this proceeding.

Q. Okay.

Dr. Thompson, are you licensed as a nuclear power plant operator?

A. No.
Q. Have you ever been licensed as a nuclear power plant operator?
A. No.
Q. Have you been trained to operate a nuclear power plant?
A. No.
Q. Have you been an engineer at a nuclear power plant?
A. No.
Q. Have you ever implemented procedures at a nuclear power plant?
A. No.
Q. Have you ever written procedures for a nuclear power plant?
A. No.
Q. Have you ever worked at a nuclear power plant?
A. No.
Q. Are you an expert in nuclear power plant operations?
A. No.
Let me -- let me correct that frame.

I have performed studies and presented testimony relating to the safety of nuclear facilities, including nuclear power plants; and in the course of those studies and preparing those testimonies, I have become expert in operational matters pertinent to the analyses and testimony. So in that limited sense, I am an expert in operations. It's a very circumscribed sense.

Q. Okay. Could you define what those areas are that you got the limited expertise in?

A. Let's take the present proceeding and Contention 2. I'm now familiar in a general sense with the configuration of the Harris Fuel Building and its equipment, and in a general sense, with the procedures used to manage fuel. I may acquire additional knowledge on these matters prior to the filing.

Q. You say you're familiar in a general sense.

MS. CURRAN: Excuse me. Before we go

on with the next question, I'd like to take a short break.

DR. HOLLAWAY: I'd like to finish the next couple questions that go directly to the question that he just responded to and I'd be happy to take a break, if that's okay.

MS. CURRAN: Okay.

Q. You said you're familiar in a general sense with the equipment at the Harris plant. What is that familiarity based on?

A. Based on -- I think I said the fuel handling building.

Q. Fuel handling building.

A. To date, that's based on review of the FSAR and other documents provided by CP&L and deciphers of yesterday.

Q. Okay. When you state --

A. -- and --

Q. Oh.

A. Correction -- and with some additional information obtained from the deposition yesterday of Mr. Devoe.

Q. Okay.

You state you're familiar in a general sense with the procedures for the fuel handling building. What's that based on?

A. Again, the same data source that I just described.

Q. Okay.

A. Data set.

Q. Your familiarity is just in a general sense, it is not from actual application?

A. That's correct. Nor would I claim to be familiar with all of the procedures used in fuel management at Harris.

Q. Okay. And even the ones that you've read or heard about, you have not actually applied yourself.

A. Correct, correct.

Q. Have you seen them applied?

A. No.

Q. Okay.

DR. HOLLAWAY: Diane, if you'd like to take a break, it will be fine.

MS. CURRAN: Okay.
DR. HOLLAWAY: How long do you want?
MS. CURRAN: Five minutes.
(Thereupon, a break was taken at 10:05 AM, with proceedings recommencing at 10:12 AM)
THE WITNESS: I'd like to clarify one of my previous statements. Is that okay?
DR. HOLLAWAY: Yes; go ahead.
THE WITNESS: You asked about my expertise in nuclear plant operations.
DR. HOLLAWAY: Yes.
THE WITNESS: And I stated that I have performed many studies and presented numerous pieces of testimony pertaining to the safety of nuclear facilities. This goes back into the 1970's. So I've become familiar with details of numerous facilities, nuclear power plants and other nuclear facilities, in several countries. And I have always taken pains to acquire the necessary familiarity with the details of the design and operation of each

facility in order to support whatever claim I made in my study or testimony.

DR. HOLLAWAY: Okay.

THE WITNESS: And that's typically not the same as the -- as the level of operational familiarity that one would require as an operator or manager of such a facility. It's a sufficiency of knowledge and expertise to support whatever claim about safety is made in the study or testimony.

And in this proceeding, I will expect to meet the same standard, that any claim that I make will be supported by sufficient expertise and familiarity with the design and procedures and operational characteristics of the Harris plant.

DR. HOLLAWAY: Okay.

Q. Your ability to speak on these issues I gather would depend on what the specific issue was?

A. I -- yes, with the clarification that I have on various occasions become --

acquired knowledge and expertise that I didn't -- did not possess up to that point --
Q. Okay.
A. -- in the realm of nuclear safety.
Q. Your familiarity with design and operations of a facility, outside of your description of time in the fuel handling building, would be based on reports you've read, documents you've read; is that correct?
A. And on applications of general physical principles.
Q. Okay. When you say "application of general physical principles," you're talking about theoretical application, not physically doing things, is that correct, yourself physically doing things?
A. I -- yes.
Q. Okay. And you say your expertise would not be the same as an operator or manager of a nuclear power plant. I presume that would include workers, technicians,

et cetera that are actually working at the facility.

A. Yes.

Each -- each such person has a particular realm of expertise, and there's only so much you can do in one life.

But I emphasize that I'm always very careful to support my claims and findings with knowledge about the underlying -- about relevant matters underlying those findings.

Q. That's certainly laudable.

How much time did you spend in the Harris Fuel Handling Building?

A. The site visit lasted about two hours, I recall; so maybe an hour in the building.

Q. Okay. Does that hour in the building make you an expert on the fuel handling building?

A. It mostly confirmed the general understanding I obtained from the FSAR.

Q. Okay; layout of where things were, et cetera.

A. Right.
Q. Okay. Have you been in other fuel handling buildings at other facilities?
A. Darlington; Main Yankee; Dukovany; and TMI, Unit 2.
Q. Where is the Darlington plant located?
A. Canada, in the province of Ontario.
Q. Okay. Is that a pressurized water reactor like Harris?
A. No.
Q. TMI, Unit 2; when were you there?
A. In the '79-80 period. I don't recall exactly. 1- -- 1980.
Q. It was after 1979.
A. Yeah.
Q. What type of reactor is Main Yankee?
A. PW- -- it -- I don't recall the vendor.
Q. And what were you doing in the fuel handling building there and for how long?
A. It was a site visit in connection with an intervention by the State of Maine.
Q. What year was that?
A. I think 1981.

Q. How long were you in that fuel handling building?
A. Maybe an hour.
Q. Dukovany; what type of reactor is that?
A. Czech Republic, for PWR units, Russian design.
Q. Russian design?
A. Soviet design.
Q. Okay. Is there an acronym that that goes by?
A. The -- the Russian for PWR is VVR.
Q. VVR?
A. Any pressurized water reactor.
Q. Okay.
What were you doing in the fuel handling building there?
A. I was representing the investor, Vienna, which in turn represented the Chancellor's Office of Austria, which was concerned about safety of fuel management at Dukovany, which is a neighboring country.
Q. What year were you there?
A. 1992.

Q. How long were you in the fuel handling building?
A. In about an hour.
Q. Okay.
You mention that part of your expertise is based on sitting in on Mr. Devoe's deposition yesterday; is that correct?
A. That's a contribution to it, yes.
Q. Okay.
A. The contribution to my knowledge, rather than expertise.
Q. Very good. How long were you in that deposition?
A. I'd guess about two hours.
Q. And did what you learned in Mr. Devoe's deposition substantially increase your knowledge on these issues?
A. No; it was a comparatively minor increase in knowledge. There were lots of loose ends left unresolved.
Q. Can you approximate, I guess percentage-wise? Is it, like, a fifty

percent increase in knowledge?
A. Oh, no; much less.
Q. One percent?
A. Less.
Q. Less than one percent?
A. Hard -- hard to say, but small. I --
Q. Okay. I mean --
A. It's not a matter that's susceptible to numerical estimate.
Q. But it's less than fifty percent?
A. Yes.
Q. Okay; less than twenty-five percent?
A. Probably, but I wouldn't give a number on that.
Q. Okay.
You have stated that you will address and do understand assumptions that go into criticality analysis.
A. Correct.
Q. Okay. Even if you don't actually do the criticality analysis yourself --
A. Correct.
Q. -- the assumptions you can address.

A. Correct.
Q. Okay.
Referring to your curriculum vitae, which is a lot of pages, on page 1 it addresses sponsors and tasks.
A. Correct.
Q. Aside from the Orange County, North Carolina, which I understand to be the present proceeding, which of these dealt with your evaluation of assumptions used in criticality analysis?
A. None of these so far.
Q. Okay.
On page 4 your CV lists publications. Aside from the first one, which is this proceeding, which of these publications address assumptions used in criticality analysis?
A. None so far.
Q. On page 8 there are expert presentations and testimony?
A. Correct.
Q. Which of these address assumptions used in

criticality analysis?
A. None.
Q. Okay.
Can you explain to me how criticality is controlled for fission reactor fuel in a spent fuel pool?
A. It can be controlled by the spacing of the fuel assemblies; by the placement of neutron-absorbing material, such as boral, between fuel assemblies; by the addition of boron to the water surrounding the fuel assemblies; and by confining placement of fuel assemblies to those which meet some specified combination of enrichment and burn-up. Those are four possible options for controlling criticality in fuel that is placed in a rack.
Q. Okay. Can you describe for me the history of development of criticality control methods for spent fuel pools?
A. In the early years of United States nuclear industry, pools employed low-density racks; and the spacing in

those racks was sufficient to preclude criticality without any other provision. As time went by and the inventory of spent fuel increased at power stations, the racks were reconfigured to bring the assemblies closer together. That -- to -- that created the potential for criticality, which was first addressed by the introduction of neutron-absorbing materials and placed between fuel assemblies, and also by the introduction of boron into the water, and more recently -- and it appears to me that it's basically an issue of the '90's -- by reliance upon restrictions of burn-up and enrichment.

So in the present state of the U.S. nuclear industry, some plants rely on all four measures in routine operation, some rely on less than all four.

And my understanding is that in this application for Pools C and D, CPL [sic] tends to rely upon three of those four

RECEIVED
DEC 03 19

WITNESS CERTIFICATION

I, GORDON THOMPSON, PH.D., do hereby certify:

That I have read and examined the contents of the foregoing two hundred and seven (207) pages of record of testimony as given by me at the time and place herein aforementioned;

And that to the best of my knowledge and belief, the foregoing two hundred and seven (207) pages are a complete and accurate record of all of the testimony given by me at said time, except as to where noted on the attached errata addendum.

G.R. Thompson 11/30/99
____

* * * * *

Sworn to and subscribed before me on the ______ day of ____________ 1999

____
Notary Public

My Commission Expires: ________________

CORRECTIONS

JOB #: MR4323X
TYPE: MR
PROOF: MR
VOL. #:
DATE TAKEN: 10-21-99

Gordon Thompson, THE WITNESS HEREIN, SUGGESTS THAT THE FOLLOWING CHANGES BE MADE IN THE TRANSCRIPT OF HIS/HER DEPOSITION IN ORDER TO MORE ACCURATELY REFLECT HIS/HER INTENDED TESTIMONY:

PAGE	LINE	READS	SHOULD READ
12	14	for a document	or a document
16	18	phase, two technical	phase with two technica
18	14-15	consultants, Institute's	consultants. The Institut
		or employees'	employs consultants
		consultants, from	from
23	12	Codes that are	Codes are
23	14	correspondence, response	correspondence, the
		for the request	response to the request
24	11	on the line	underlying the
34	6	Main	Maine
34	15	1 - -- 1980.	I think perhaps 1980.
34	18	Main	Maine
34	19	PW	PWR

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
35	7	for PWR	four PWR
35	19	investor, Vienna	University of Vienna
35	23	is a neighboring	is in a neighboring
36	5	In about an	About an
36	13	The contribution	A contribution
39	23	of United	of the United
41	7	understanding,	understanding is
42	16-17	can sort of guard	Can regard as
		as low-density,	low-density, in
		open	an open
43	19	facility at which	facility which
47	7	limit the	limiting the

5/92

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
47	22	is very important	is a very important
		to a regulatory	regulatory
48	25	criticality instant	criticality incident
49	3	making two	moving two
53	7	taking credit	taking of credit
54	10	burn-up enrichment	burn-up and
		and enrichment	enrichment
54	15	machinery or the	machinery or the
			provision of
60	10	and now brief	in our brief
60	21	than were on	than are boron
62	13	cycle a fission	cycle of fission
62	14	born and causes	born until it causes

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
62	15-16	that --because it's of interest	that is of interest
62	23	the primer that	the parameter that
63	10	may -- is -- is	may approach
63	16-17	they tend to be self -- tend to be -- tend to be	the reaction tends to be
63	19	analyses of this	analyses within ~~which~~ this
64	22	in that finite	in a finite
64	23	neurons	neutrons
65	7	fission	fissile
66	15	fission	fissions
66	21	uranium	linear

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
69	18-19	viewgraphs ~~are~~	viewgraphs drew
		drawn in NRC	on NRC
71	12	the case	this case
71	16	the safety	their safety
74	12	here. In	here in
74	13	regard -- Contention	regard to Contention
74	14	at this point would	of this point
		be on my	are beyond my
75	14	by a Commission	by the Commission
75	16	meets	meet
76	25	improved	approved
77	3-4	it goes to the design	
		-- it can be --	

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
77	7	and that	and there
77	9-10	requirement at --	requirement in terms
		in terms of site-	of site-specific
		specific requirement,	requirements and
77	21	cases	casks
88	8	by -- in	in
88	9	cask,	cask, a
88	14	You turn	If you turn
88	15	attachment,	attachment
89	11	pocket	onset
89	14	The analysis -- numerous	Numerous
89	21	at least be limited	be limited to much less
		to that -- all	than that in all

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
90	11	fueling	filling
91	14	clouding	cladding
91	16	steam is occurring	steam - zirconium or
		or if occurring	air - zirconium reaction
96	7	shared	showed
100	14	potential for	potential for a
105	11	initiate	initiated
105	15	Tandy	Sandia
125	16	frames had, the	framers had for the
125	21	serious that you	seriously you
125	22	as my	of my
126	23	one side there's	the other side is
131	7	assume that	assume as

C O R R E C T I O N S

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PAGE	LINE	READS	SHOULD READ
131	8	done -- in	done in
131	18	for -- in C	for pools C
134	20	couple	couple of
134	21-22	contention on a	contention and a
134	23	using its words	using words
148	16	different	driven
148	17-18	power facilities	power licensees
148	19	incurred in	of
148	21	occurrence	occupancy
148	23	the control	criticality control
148	24	frame by the	framing of
150	14	geometric and	geometrically
156	19	assign the	assign

Brief Exhibit 2

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY & LICENSING BOARD

- - - - - - - - - - - - - - - - - - - -x

In The Matter Of: :

CAROLINA POWER & LIGHT COMPANY, :

(Shearon Harris Nuclear Power :

Plant) :

- - - - - - - - - - - - - - - - - - - -x

Washington, D.C.

Thursday, October 14, 1999

Deposition of DAVID A. LOCHBAUM, called for examination, pursuant to notice, at 10:10 a.m., at the offices of Shaw Pittman, 2300 N Street, NW, Third Floor, Washington, D.C., before Mario A. Rodriguez, a notary public in and for the District of Columbia, when were present on behalf of the respective parties:

APPEARANCES:

On behalf of Carolina Power & Light Company:

JOHN H. O'NEILL, JR., ESQ.
WILLIAM R. HOLLAWAY, ESQ.
Shaw Pittman
2300 N Street, NW
Washington, D.C. 20037
(202) 663-8000

On behalf of Nuclear Regulatory Commission:

SUSAN L. UTTAL, ESQ.
U.S. Nuclear Regulatory Commission
Washington, D.C 20444
(301) 415-1582

On Behalf of the Board of Orange County Commissioners:

DIANE CURRAN, ESQ.
Harmon, Curran, Spielberg & Eisenberg, LLP
1726 M Street, NW

Suite 600
Washington, D.C.
(202) 328-3500

ALSO PRESENT:

JAMES A. DAVIS, Materials Engineer
KENNETH C. HECK, Operations Engineer
U.S. Nuclear Regulatory COmmission

PROCEEDINGS

Whereupon,

DAVID LOCHBAUM,

a witness, was called for examination by counsel and, having been first duly sworn, was examined and testified as follows:

MR. O'NEILL: First instructions to the court reporter: To transcribe everything during the deposition except during breaks or mutual off-the-record discussions when nothing should be transcribed.

Interrupt when necessary to clear up any doubts about a question or an answer that you have since what you transcribe is what's important.

Please transcribe the attendances and the exists and entrances of any individual during the deposition.

And we've already introduced ourselves prior to going on the record and we note that you have all of the individuals for the record at the moment.

I'll ask you to mark all exhibits prior to

And then I went to work at the Grand Gulf Plant during its initial construction.

Q And did you have responsibilities as a construction engineer or on the operations side?

A It was on the operations side.

Q As a start-up engineer?

A As a start-up engineer, yes.

Q Have you had any experience as a construction engineer?

A No, I have not.

Q Have you had any responsibility for welding at a nuclear power plant?

A No, I have not.

Q Have you had any responsibility for construction quality assurance or quality control at a nuclear power plant?

A I have in a standpoint -- I worked for a brief while for General Electric, and one of the assignments was at the Grand Gulf Nuclear Power Station. The plant owner or plant licensee asked me to go through the nonconformance reports that were written against GE products and services during the tail end of construction to ensure that they were resolved, dispositioned, make sure that there weren't any that were missed.

A lot of those involved -- they used a head -- bolt, take out the main steamline plugs and missed, they hit the vessel instead of the plug. So a lot of these were to ensure that, you know, equipment was either repaired, reworked or accepted as is or there was some kind of disposition.

So I had to review hundreds of those things and track them down.

Q Have you been responsible for writing or modifying QA procedures?

A Well, also at the Grand Gulf plant for GE, one of the things I had was -- the independent safety engineering group was being formed in response to NUREG 0656 -- I think it's 0646 or 0656. I can't recall offhand. But you are required to have an independent safety and engineering group. One of its responsibilities is to periodically verify the adequacy of the on-site QA/QC group.

So I wrote the procedures for the

independent safety and engineering group to perform that audit function.

Q By the way, have you reviewed the QA procedures that are being used to commission the spent fuel pool cooling system for unit 2 for pools C and D?

A If they were in the application, I did. There were also some documents like that that we've requested that I have not yet reviewed.

Q Do you know what NDE stands for?

A Yes, I do.

Q What is it?

A Non-destructive examination.

Q Have you been responsible for NDE at any nuclear plant in any way?

A Not in a traditional sense. I've done a lot of examination that didn't result in destruction, but not NDE as you use it.

Q Have you ever qualified as an NDE examiner?

A No, I have not.

Q Have you been responsible for NDE

examiners at any nuclear plant?

A No, I have not.

Q Have you ever welded materials together?

A No, I have not.

Q And I believe you indicated earlier you have not had any responsibility as a welding engineer.

A That's correct.

Q Have you ever been responsible for the QA or QC inspectors at a nuclear plant?

A No, I have not.

Q Have you serviced on any ASME code committees?

A No, I have not.

Q Are you an expert in material science?

A No.

Q Are you an expert in corrosion of materials at a nuclear power plant?

A No, I'm not.

Q Are you an expert in stress analysis?

A No, I'm not.

Q Are you an expert in failure analysis?

A I've been trained in root cause analysis, so with that slice of it, that's a smaller subset than failure analysis in a broad sense.

Q Tell me what your experience has been in failure analysis.

A As an STA, shift technical advisor at Browns Ferry, part of your job is to figure out what happens: Do the post trip analysis, do the preliminary determination when a piece of equipment fails as to what caused it and what happened.

So in addition, for certain things we were required to write the licensee event reports that were later submitted to the NRC. That involved -- sometimes if it was an area within my system responsibility, I would do the failure analysis, find out what happened.

For the areas that were not within my responsibility, as the STA I was responsible for working with whoever was responsible to identify what the failure was and get that information in to the licensee event report or the post trip report or whatever the proper document was.

Q Are you an expert in the causes of degradation of stainless steels?
A No, I'm not.
Q Are you an expert in probability and statistics as it applies to engineering design?
A No.
Q What is the diameter of the piping at the union 2 spent fuel pool cooling system?
A I don't know. I doubt it would be all the same diameter.
Q What are the diameters if they are not all the same?
A I don't know.
Q Any idea?
A No.
Q What is the thickness of the piping at the Harris spent fuel pool cooling system for unit 2?
A I don't know.
Q What is the materials of the piping for the spent fuel pool cooling system for unit 2?
A Some of it, if not all of it, is stainless steel.

to contention TC3."

Interrogatory number 3 requests identification of individuals who are experts and expected to provide sworn affidavits and declarations for the written filing.

On what areas as an expert will you provide written sworn testimony?

A Well, the snide answer would be the answers -- the areas covered under technical contention number 3, which were the quality assurance and the readiness of the spent fuel pool cooling system to be used.

Q I understand that. But we've established some areas that are related that you are not an expert. So now I want you to tell me what areas relating to contention 3 that you consider yourself an expert and, therefore, qualified to give an expert opinion.

A Go back to the areas we just went through with the yeses and nos. The areas of quality assurance, where this plant has completed -- has all the documentation necessary and the work necessary to put this systems in service would be the areas I would be looking at in providing an affidavit or a declaration or some document.

Q But you will not be taking a position on corrosion, I understand, since you're not an expert in corrosion?

A I will not be saying whether a component is corroded or not because I don't have the ability to do that.

I can look at nothing and determine it wasn't addressed and that the level showing that this is -- all the bases have been covered, I can determine whether that has been done or not.

So I think there is a distinction -- those are my boundaries as far as what I --

Q And you certainly will not be giving an opinion on welding, for example?

A I will not be saying, looking at some of the information we just looked at in discovery and saying, you know, CP&L says this weld is good and I'll say, no, this weld is bad. I won't venture anything like that.

But if they have not a process in place that demonstrates that all the welds are adequate, I could point out flaws or gaps in that process.

MR. O'NEILL: Off the record.

(Discussion off the record.)

THE WITNESS: Before we resume, could I clarify two things that we talked about earlier today?

MR. O'NEILL: Sure.

THE WITNESS: One of them had to do with the retention and the expert witness part.

BY MR. O'NEILL:

Q Yes.

A I considered myself retained in the same capacity as Gordon Thompson, the difference being -- there's two differences. One is I'm not getting compensated, whereas Mr. Thompson is, at least for his travel, perhaps for his time, I don't know. But UCS is a public interest group. We do things like this. That's how we get money from donors and stuff.

So I don't want to get UCS in a position

A In this application I do. It's used in a number of places, and I can't swear that I understand where 50.55a is used everywhere else, but in this application I'd say yes.

Q Have you ever prepared a 50.55a application in your work for a utility to obtain an exemption to a code requirement?

A I've not prepared one. I've been the reviewer for plans when I worked in licensing groups.

Q Okay. What is the requirement, the code requirement that the 50.55a plan addresses? The code requirement.

A You're talking about the ASME code requirement?

Q Correct.

A I don't recall offhand what the wording of that code requirement is.

Q If you don't recall the wording, do you understand what requirement the 50.55a plan addresses?

A The purpose of the code is to ensure, or the function of the code is to ensure that there is a certain quality standard that are met prior to the use of any safety-related system.

Q Okay. Now, what specifically, very specifically, does CP&L say it cannot meet and, therefore, requires an alternative plan?

A It cannot meet the quality assurance documentation of the welds and the construction of the spent fuel cooling system on unit 2. It lost -- some of the records were destroyed -- were inadvertently destroyed and so on. It lacks that pedigree.

Q So it lacks some records for certain welds. Anything else?

A Well, the commissioning plan, not the alternative plan, there were also some things that were not yet installed and they had to go out and verify that the installation was complete.

Q But that's not part of the 50.55a plan, is it?

A That's correct.

Q Because the 50.55a plan only goes to what knowledge and understanding of this process which you are the expert on?

A I don't believe so, no.

Q Okay. For example, the 50.55a plan does not address the heat exchangers, does it?

A That's correct.

Q Why doesn't it address the heat exchangers?

A It's not required to address the heat exchangers.

Q Because the heat exchangers meet all the code requirements; is that not correct?

A I'm not going to swear to that, no.

Q Okay. But you have no reason to believe that the heat exchangers don't meet the code requirements?

A I've never looked at that question, so I'm not going to say yes or no.

Q It's not part of this contention, is it?

A It is not part of this contention. That I can answer.

Q Do you happen to know how the heat

exchangers were stored?

A I do not happen to know how the heat exchangers were stored.

Q But the heat exchangers can be inspected carefully to ensure that, A, they meet the code requirements and, secondly, that they haven't deteriorated, can they not?

A Yes, the heat exchangers can be inspected to ensure that, A, they met all the code and, B, that they haven't deteriorated, yes.

Q Similarly, the pumps can be inspected, can they not?

A The pumps can be similarly inspected, yes.

Q The piping that is accessible and not embedded in concrete can also be inspected, can it not?

A The piping -- even the embedded piping can be inspected, yes.

Q Okay. But the piping that is accessible can be inspected both with respect to the ID and the OD, can it not?

A Would you --

Q The piping that is accessible that is not embedded in concrete can be inspected from both the ID and from the OD, can it not? Internal diameter, outside diameter.

A Yes, it could.

Q Okay. With respect to the welds and the accessible piping, even if the weld data reports are missing, they can be recreated, can they not?

A I'm not sure that all the weld records can be recreated. There are certain -- no, you cannot recreate all the original weld data. No, you can't.

Q Is it your position that you cannot recreate a weld data record for welds that can be inspected and their pedigree can be verified both by inspection external and internal?

A Part of the original welds records, data records, includes the welder's name and qualifications, and it's hard to do that by inspection 18 years later, so data like that is not going to be able to --

Q Isn't it true that there is a welder symbol that is inscribed next to each of the welds?

A There is. I don't know offhand if the cross-reference between those symbols and the welder's name is part of the records that were retained or part of the records that were discarded.

Q And you didn't review those records that were provided at CP&L's offices with respect to all of the welds and all of the piping and all of the QA records that have been amassed relating to that piping?

A I believe I answered earlier, we requested some documents. I haven't had a chance to review those documents. So I stand by that previous answer.

Q This contention, however, does not address the welds with respect to the accessible piping, does it?

A No, it does not.

Q And, indeed, the 50.55a application doesn't address the welds with respect to the accessible piping, does it?

A That is correct.

Q The only thing that this contention

addresses, is it not true, is the embedded piping and embedded welds?

A The way it's worded, that's correct.

Q Well, that's what we're talking about is the way it's worded, right? I mean, that's the issue.

A That's correct.

Q In fact, this was recrafted to make sure that the issue was clarified after the prehearing conference, and this pleading is, indeed, Orange County's recrafting of the contention?

A I understand that.

Q In the April 7th, 1999, presentation that you made to the commissioners and the public -- if you will look at Exhibit 4. And you didn't number your pages here, but if you look at --

A Yes, I did.

Q I'm sorry. Slide 7. Yes, you did. Thank you. Slide 7. The last bullet says, "But the alternative plan covers the system in 1983, not how the intervening 15 years (of rust and neglect?) have affected it."

CERTIFICATE OF DEPONENT

I, DAVID A. LOCHBAUM, 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.

David A. Lochbaum ______________________

11-30-99 ______________________

DATE

Sworn and subscribed to before me, this the _________ day of _______________, 19____.

____

NOTARY PUBLIC IN AND FOR

My commission expires:

ERRATA SHEET

Re: Carolina Power & Light Company, (Shearon Harris Nuclear Power Plant)

Case No.:

Date Taken: Thursday, October 14, 1999

Deposition of: DAVID A. LOCHBAUM

I hereby certify that I have read my deposition and that it is accurate, with the corrections listed below:

Page	Line	As Transcribed	Change To:
13	22	NC Warren	NC-WARN
20	3	NC Warren	NC-WARN
20	13	NC Warren	NC-WARN
27	19	evac	HVAC
32	2	state class 1E	safety class 1E
32	22	diesel generates	diesel generators
53	3	SLB	ASLB
53	4	SLB	ASLB
53	11	SLB	ASLB
53	17	SLB	ASLB
53	21	SLB	ASLB
54	12	SLB	ASLB
54	14	SLB	ASLB
61	21	SLB	ASLB
62	2	Yankee Row	Yankee Rowe
62	6	Yankee Row	Yankee Rowe
62	15	Yankee Row	Yankee Rowe
65	9	SLB	ASLB
126	14	IMPO	INPO
150	22	reactor material	radioactive material
165	18	end-stamped	N-stamped

11-30-99
Date

David A. Lochbaum
Signature of Deponent

NOTE: If there are no corrections, write "None" above. Use additional pages if necessary. Be sure you have dated the Errata Sheet.

NOTE: Page 152, line 22 through Page 154, line 14 is A duplicate (unnecessary) of Page 151, line 7 through Page 152, line 21.

ERRATA SHEET

Re: Carolina Power & Light Company, (Shearon Harris Nuclear Power Plant)

Case No.:

Date Taken: Thursday, October 14, 1999

Deposition of: DAVID A. LOCHBAUM

Page	Line	As Transcribed	Change To:
184	3	hydrostatic Kelly	hydrostatically
222	14	dry cast storage	dry cask storage
222	16	dry cast	dry cask

DAL

11-30-99

DOCKETED
USNRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'00 JAN -6 A9:04

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

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney herewith enters an appearance in the above-captioned matter. In accordance with 10 C.F.R. § 2.713(a), the following information is provided:

Name: Robert M. Weisman

Address: U.S. Nuclear Regulatory Commission
Office of the General Counsel
Washington, D.C. 20555

Telephone Number: (301) 415-1696
Facsimile: (301) 415-3725

Internet Address: RMW@NRC.GOV

Admissions: Supreme Court of the State of Oklahoma (I.D. # 011006)

Name of Party: NRC Staff

Respectfully submitted

Robert M. Weisman

Robert M. Weisman
Counsel for NRC Staff

Dated at Rockville, Maryland
this 4th day of January, 2000