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June 4, 2002

Mark Langer, Clerk
U.S. Court of Appeals
For the District of Columbia Circuit
3rd and Constitution Avenues N.W.
Washington, D.C. 20001

SUBJECT: *Orange County v. NRC, Nos. 01-1073 and 01-1246*

Dear Mr. Langer,

Enclosed please find ten copies of the deferred Joint Appendix and the exhibits in this case. I am also enclosing the original and four copies of a consent motion to file the Joint Appendix one day late. (It was due on June 3).

Copies of these filings have been served on the parties to the case.

Sincerely,

Diane Curran

Encl: As Stated
Cc. w/Encl.: Service list

Template 06C002

SECY-02 06C01

UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

ORANGE COUNTY, NORTH CAROLINA, Petitioner, v. UNITED STATES NUCLEAR REGULATORY COMMISSION and the UNITED STATES OF AMERICA, Respondents, CAROLINA POWER & LIGHT CO., Intervenor.	)	Nos. 01-1073, 01-1246 (Consolidated)

CONSENT MOTION FOR EXTENSION OF TIME TO FILE DEFERRRED JOINT APPENDIX

The Board of Commissioners of Orange County, North Carolina ("Orange County"), hereby moves for a one-day extension of time, from June 3 to June 4, 2002, to file the deferred Joint Appendix in the above-captioned case. Copies of the Joint Appendix are being filed with this motion. The extension was necessary because counsel for Orange County had difficulty reaching counsel for the other parties in order to coordinate the preparation of the Joint Appendix.

Both the respondent, the U.S. Nuclear Regulatory Commission, and the intervenor, Carolina Power & Light Company, have consented to this motion.

Respectfully submitted,

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June 4, 2002

UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

ORANGE COUNTY, NORTH CAROLINA, Petitioner, v. UNITED STATES NUCLEAR REGULATORY COMMISSION and the UNITED STATES OF AMERICA, Respondents CAROLINA POWER & LIGHT Intervenor-Respondents	Nos. 01-1073, 01-1246 (Consolidated)

CERTIFICATE OF SERVICE

I certify that on June 4, 2002, copies of the Joint Appendix and Exhibits, and copies of a motion for leave to file same out of time, were served on the Court and on the following parties by hand delivery:

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CASE SCHEDULED FOR ORAL ARGUMENT SEPTEMBER 5, 2002

In the
United States Court of Appeals
For the District of Columbia Circuit

Nos. 01-1073 and 01-1246 (Consolidated)

ORANGE COUNTY, NORTH CAROLINA, *Petitioner*
v.
UNITED STATES NUCLEAR REGULATORY COMMISSION
And the UNITED STATES OF AMERICA, *Respondents*
CAROLINA POWER & LIGHT COMPANY, *Intervenor-Respondent*

PETITION TO REVIEW A FINAL DECISION OF THE
U.S. NUCLEAR REGULATORY COMMISSION

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Dated: June 4, 2002

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Richard A. Meserve, Chairman
Greta Joy Dicus
Nils J. Diaz
Edward McGaffigan, Jr.
Jeffrey S. Merrifield

In the Matter of **Docket No. 50-400-LA**

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant) **May 10, 2001**

The Commission denies Intervenor's petition for review of three Licensing Board decisions (LBP-00-12, LBP-00-19, and LBP-01-9) that rejected challenges to issuance of a license amendment for spent fuel pool expansion. The Commission also denies Intervenor's request for a stay.

RULES OF PRACTICE: APPELLATE REVIEW

The Board's decisions for the most part rest on its own carefully rendered fact findings, an area where we repeatedly have declined to second-guess plausible Board decisions. *See, e.g., Hydro Resources, Inc.* (P.O. Box 15910, Rio Rancho, NM 87174), CLI-01-4, 53 NRC 31, 45 (2001); *Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), CLI-98-3, 47 NRC 77, 93 (1998); *Kenneth G. Pierce* (Shorewood, Illinois), CLI-95-6, 41 NRC 381, 382 (1995).

RULES OF PRACTICE: APPELLATE REVIEW

On a petition for review, Petitioner must adequately call the Commission's attention to claimed errors in the Board's approach. Here, Petitioner has submitted a complex set of pleadings that includes numerous detailed footnotes, attachments,

and incorporations by reference. We deem waived any arguments not raised before the Board or not clearly articulated in the petition for review. *See Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 46; *Commonwealth Edison Co.* (Zion Nuclear Power Station, Units 1 and 2), CLI-99-4, 49 NRC 185, 194 (1999); *Curators of the University of Missouri*, CLI-95-1, 41 NRC 71, 132 n.81 (1995).

10 C.F.R. § 2.206

PUBLIC PETITIONS

Safety questions not properly raised in an adjudication may nonetheless be suitable for NRC consideration under its public petitioning process, 10 C.F.R. § 2.206. *See Power Authority of the State of New York* (James A. FitzPatrick Nuclear Power Plant; Indian Point, Unit 3), CLI-00-22, 52 NRC 266, 311 (2000); *International Uranium (USA) Corp.* (Receipt of Material from Tonawanda, New York), CLI-98-23, 48 NRC 259, 265-66 (1998).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

LICENSING BOARDS: AUTHORITY

Our Subpart K process empowers a licensing board to *resolve* fact questions, when it can do so accurately, at the abbreviated hearing stage.

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS; HEARING ON CONTENTIONS

Subpart K establishes a two-part test to determine whether a full evidentiary hearing is warranted: (1) there must be a genuine and substantial dispute of fact "which can only be resolved with sufficient accuracy" by a further adjudicatory hearing; and (2) the Commission's decision "is likely to depend in whole or in part on the resolution of that dispute." *See* 10 C.F.R. § 2.1115(b).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

REGULATIONS: INTERPRETATION (10 C.F.R. PART 2, SUBPART K)

NUCLEAR WASTE POLICY ACT: HEARING REQUIREMENTS

"In promulgating section 2.1115(b) of Subpart K, we used the same test described in the Nuclear Waste Policy Act of 1983 ["NWPA"] at 42 U.S.C. § 10154(b)(1). We noted that

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

See id. at 26 n.5, *quoting* Final Rule, "Hybrid Hearing Procedures for Expansion of Spent Nuclear Fuel Storage Capacity at Civilian Nuclear Power Reactors," 50 Fed. Reg. 41,662, 41,667 (Oct. 15, 1985).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

NUCLEAR WASTE POLICY ACT: HEARING REQUIREMENTS

Subpart K derives from the NWPA, where Congress called on the Commission to "encourage and expedite" onsite spent fuel storage. *See* 42 U.S.C. § 10151(a)(2). To help accomplish this goal, the NWPA required the Commission, "at the request of any party," to employ an abbreviated hearing process — i.e., discovery, written submissions, and oral argument. *See* 42 U.S.C. § 10154. The NWPA authorized the Commission to convene additional "adjudicatory" hearings "only" where critical fact questions could not otherwise be answered "with sufficient accuracy." *See* 42 U.S.C. § 10154(b)(1)(A).

RULES OF PRACTICE: SUMMARY DISPOSITION

Our rules long have allowed summary disposition in cases where "there is no genuine issue as to any material fact" and where "the moving party is entitled to a decision as a matter of law." *See* 10 C.F.R. § 2.749(d); *cf.* Fed. R. Civ. P. 56 (judicial summary judgment rule).

RULES OF PRACTICE: HEARING PROCEDURES IN SPENT FUEL POOL EXPANSION PROCEEDINGS

REGULATIONS: INTERPRETATION (10 C.F.R. PART 2, SUBPART K)

RULES OF PRACTICE: SUMMARY DISPOSITION

It seems unlikely to us that Congress intended the Commission to enact Subpart K simply to replicate the NRC's existing summary disposition practice. Congress "cannot be presumed to do a futile thing." *Halverson v. Slater,* 129 F.3d 180, 184 (D.C. Cir. 1997). *Accord Independent Insurance Agents of America, Inc. v. Hawke,* 211 F.3d 638, 643 (D.C. Cir. 2000). Hence, to give real-world meaning to Subpart K's abbreviated hearing process, we construe Subpart K to extend beyond the NRC's pre-existing summary disposition practice. Unlike our summary disposition rule, which requires an additional evidentiary hearing whenever a licensing board finds, based on the papers filed, that there remains a genuine issue of material fact, Subpart K's procedure authorizes the board to *resolve* disputed facts based on the evidentiary record made in the abbreviated hearing, without convening a full evidentiary hearing, if the board can do so with "sufficient accuracy."

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

REGULATIONS: INTERPRETATION (10 C.F.R. PART 2, SUBPART K)

LICENSING BOARDS: RESOLUTION OF ISSUES

Subpart K directs the Board to "[*d*]*ispose* of any *issues* of law or *fact* not designated for resolution in an adjudicatory hearing." *See* 10 C.F.R. § 2.1115(a)(2) (emphasis added). "Issues" are, by definition, points of debate or dispute. To "dispose" of issues, a board must *resolve* them. To move from Subpart K's abbreviated hearing stage to an additional evidentiary hearing, a licensing board must make a specific determination that issues "can *only* be resolved with sufficient accuracy" at such a hearing. *See* 10 C.F.R. § 2.1115(b)(1) (emphasis added).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

REGULATIONS: INTERPRETATION (10 C.F.R. PART 2, SUBPART K)

EVIDENCE: EXPERT WITNESSES

RULES OF PRACTICE: EXPERT WITNESSES

The Statement of Considerations for Subpart K reinforces the rule's text:

> The appropriate evidentiary weight to be given an expert's technical judgment will depend, for the most part, on the expert's testimony and professional qualifications. In some circumstances, it may be possible to make such a determination without the need for an adjudicatory hearing. The presiding officer must decide, based on the sworn testimony and sworn written submissions, whether the differing technical judgment gives rise to a genuine and substantial dispute of fact that *must* be resolved in an adjudicatory hearing.

See 50 Fed. Reg. at 41,667 (1985) (emphasis added).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

REGULATIONS: INTERPRETATION (10 C.F.R. PART 2, SUBPART K)

NUCLEAR WASTE POLICY ACT

The NWPA and our rule implementing it (Subpart K) contemplate merits rulings by licensing boards based on the parties' written submissions and oral arguments, except where a board expressly finds that "accuracy" demands a full-scale evidentiary hearing.

LICENSING BOARDS: RESOLUTION OF ISSUES

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS

Licensing boards are fully capable of making fair and reasonable merits decisions on technical issues after receiving written submissions and hearing oral arguments. The Commission is a technically oriented administrative agency, an orientation that is reflected in the makeup of its licensing boards. Most licensing boards have two, and all have at least one, technically trained member. In Subpart K cases, licensing boards are expected to assess the appropriate evidentiary weight to be given competing experts' technical judgments, as reflected in their reports and affidavits. The inquiry is similar to that performed by presiding officers in materials licensing cases, where fact disputes normally are decided "on the papers," with no live evidentiary hearing. *See, e.g., Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 45; *Curators of the University of Missouri*, CLI-95-1, 41 NRC at 118-20. The NRC's administrative judges, in other words, and the Commission itself, are accustomed to resolving technical disputes without resort to in-person testimony.

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS; WITNESSES (CREDIBILITY)

There may be issues, such as those involving witness credibility, that cannot be resolved absent face-to-face observation and assessment of the witness. Or there may be issues involving expert or other testimony where key questions require followup and dialogue to be answered "with sufficient accuracy." In these kinds of cases, Subpart K contemplates further evidentiary hearings. Many issues, however, particularly those involving competing technical or expert presentations, frequently are amenable to resolution by a licensing board based on its evaluation of the thoroughness, sophistication, accuracy, and persuasiveness of the parties' submissions.

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS; APPELLATE REVIEW

NUCLEAR WASTE POLICY ACT: HEARING REQUIREMENTS

On a case-by-case basis, we generally will defer to our licensing boards' judgment on when they will benefit from hearing live testimony and from direct questioning of experts or other witnesses. If a decision can be made judiciously on the basis of written submissions and oral argument, we expect our boards to follow the mandate of the NWPA and Subpart K to streamline spent fuel pool expansion proceedings by making the merits decision expeditiously, without additional evidentiary hearings. *See* 42 U.S.C. §§ 10151(a)(2), 10154.

RULES OF PRACTICE: APPELLATE REVIEW

LICENSING BOARDS: RESOLUTION OF ISSUES

The Commission is generally not inclined to upset the Board's fact-driven findings and conclusions, particularly where it has weighed the affidavits or submissions of technical experts. Here, in our judgment, the Board analyzed the

parties' technical submissions carefully, and made intricate and well-supported findings in a 42-page opinion. We see no basis for the Commission, on appeal, to redo the Board's work.

NEPA: REQUIREMENT FOR ENVIRONMENTAL IMPACT STATEMENT; EIS (SEVERE ACCIDENTS; SPENT FUEL POOL; RULE OF REASON)

The Commission has never determined a threshold accident probability figure for imposing the requirement of preparing an EIS. Eleven years ago, the Commission indicated that such a threshold would be "better explored outside the scope of a particular case involving only a few parties," and declined "either to endorse or reject" an Appeal Board determination that an accident probability of 10^{-4} is remote and speculative. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333, 335 (1990). In a later decision in that same proceeding, the Commission reiterated that "low probability is the key to applying NEPA's rule-of-reason test to contentions that allege that a specified accident scenario presents a significant environmental impact that must be evaluated." *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 131 (1990). Because we do not disturb the Board's finding of extremely low probability in this case, we need not decide here whether Orange County's 1.6×10^{-5} probability estimate is remote and speculative so as not to require preparation of an EIS.

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDINGS; HEARING ON CONTENTIONS; BURDEN OF GOING FORWARD

The proponent of a contention must supply, at the written submission and oral argument stages of a Subpart K proceeding, all of the facts upon which it intends to rely at the formal evidentiary hearing, should one prove necessary. *See Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 3), CLI-01-3, 53 NRC 22, 27 (2001).

RULES OF PRACTICE: HEARING PROCEEDINGS FOR SPENT FUEL POOL EXPANSION PROCEEDINGS; HEARING ON CONTENTIONS

NUCLEAR WASTE POLICY ACT

We see no basis for upsetting the Board's probability estimate or its decision against a further evidentiary hearing. Even if a further evidentiary hearing were convened, Intervenor apparently intends merely to reiterate its critique of the probabilistic risk assessment of others (the NRC Staff and the Licensee), but not to offer a fresh analysis of its own. Under these circumstances, scheduling a further hearing would serve only to delay these proceedings and increase the costs for all parties, in direct contravention of the NWPA.

RULES OF PRACTICE: APPELLATE REVIEW; CONTENTIONS (SPECIFICITY AND BASIS)

At the contentions stage of this litigation, Orange County offered no specific causes for spent fuel pool accidents other than the seven-step scenario admitted by the Board. Orange County cannot now transform vague references to potential spent fuel pool catastrophes into litigable contentions. *See Duke Energy Corp.* (Oconee Nuclear Station, Units 1, 2, and 3), CLI-99-11, 49 NRC 328, 333-35 (1999) (NRC's "strict contention rule" requires "detailed pleadings").

RULES OF PRACTICE: APPELLATE REVIEW

APPELLATE REVIEW: RAISING MATTERS FOR FIRST TIME

Orange County expressly approved the final language of its admitted environmental contention. The County should not now be heard to complain that the contention as admitted was too narrow.

QUALITY ASSURANCE/QUALITY CONTROL

RULES OF PRACTICE: APPELLATE REVIEW

The Board did not give short shrift to Intervenor's quality assurance concerns. The Board admitted the issue for hearing, allowed discovery, obtained written evidence, and heard oral argument. The Board ultimately devoted some 11 pages of its order to discussing the quality assurance issue on the merits. *See* LBP-00-12, 51 NRC at 269-80. We do not ordinarily second-guess Board fact findings, particularly those reached with this degree of care.

RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS

Intervenor maintains that the Board erred in refusing to consider its argument that the Licensee must seek a construction permit to use the piping and equipment that were abandoned in the early 1980s. The Board ruled that the construction permit claim was not a part of Intervenor's admitted contention and cannot be

admitted unless it fulfills the late-filing standards set out in 10 C.F.R. § 2.714(a). *See* LBP-00-12, 51 NRC at 281. Because Intervenor made no effort to address the late-filing standards, the Board precluded further consideration of the issue. *See id.* at 281-82. We agree with the Board. Intervenor was inexcusably late in attempting to introduce its construction permit claim.

CONSTRUCTION PERMITS: MATERIAL ALTERATION

The Board expressed skepticism that the amendment proposed by Licensee "is a 'material alteration' in the sense intended by the regulations so as to require a construction permit." *See* LBP-00-12, 51 NRC at 281-82, *citing* 10 C.F.R. § 50.92(a). Alterations of the type that require a construction permit are those that involve substantial changes that, in effect, transform the facility into something it previously was not or that introduce significant new issues relating to the nature and function of the facility. *See Portland General Electric Co.* (Trojan Nuclear Plant), LBP-77-69, 6 NRC 1179, 1183 (1977). To trigger the need for a construction permit, the change must "essentially [render] major portions of the original safety analysis for the facility inapplicable to the modified facility." *See id.*

STAYS PENDING APPELLATE REVIEW

RULES OF PRACTICE: STAY OF AGENCY ACTION (CRITERIA); STAY PENDING APPEAL

Stays pending appellate review are governed by 10 C.F.R. § 2.788. In determining whether to grant a stay, the Commission will consider:

> (1) Whether the moving party has made a strong showing that it is likely to prevail on the merits;
> (2) Whether the party will be irreparably injured unless a stay is granted;
> (3) Whether the granting of a stay would harm other parties; and
> (4) Where the public interest lies.

See 10 C.F.R. § 2.788(e).

RULES OF PRACTICE: STAY OF AGENCY ACTION; STAY PENDING APPEAL

Our decision today to deny Orange County's petition for review terminates adjudicatory proceedings before the Commission, and renders moot the County's motion for a stay pending appeal.

RULES OF PRACTICE: STAY PENDING APPEAL; STAY OF AGENCY ACTION (IRREPARABLE INJURY)

We took no action on Intervenor's stay motion during our consideration of the Intervenor's petition for review because we saw no possibility of irreparable injury. The record indicates that the injury asserted by Intervenor could not occur until at least July 2, 2001, when the Licensee expects to place spent fuel pools C and D into service following testing. Even after July 2, the additional spent fuel stored at Shearon Harris will total no more than 150 fuel elements in the short term (i.e., during 2001). Moreover, Intervenor's claim of injury — offsite radiation exposure in the event of a spent fuel pool accident — is speculative, given the small likelihood of such an accident, and does not amount to the kind of "certain and great" harm necessary for a stay. *See Cuomo v. NRC*, 772 F.2d 972, 976 (D.C. Cir. 1985); *accord Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-820, 22 NRC 743, 747-48 & n.20 (1985).

RULES OF PRACTICE: STAY PENDING APPEAL; STAY OF AGENCY ACTION (IRREPARABLE INJURY)

Of the four stay factors, "the most crucial is whether irreparable injury will be incurred by the movant absent a stay." *Alabama Power Co.* (Joseph M. Farley Nuclear Plant, Units 1 and 2), CLI-81-27, 14 NRC 795, 797 (1981). *Accord Sequoyah Fuels Corp. and General Atomics* (Gore, Oklahoma Site), CLI-94-9, 40 NRC 1, 7 (1994).

RULES OF PRACTICE: APPELLATE REVIEW; COMPLIANCE WITH COMMISSION RULES

The Commission's rule providing for review of decisions of a presiding officer plainly states that a "petition for review . . . must be no longer than ten (10) pages." *See* 10 C.F.R. § 2.786(b)(2). Orange County's petition for review, although nominally confined to 10 pages, resorts to the use of voluminous footnotes, references to multipage sections of earlier filings, and supplementation with affidavits that include additional substantive arguments. This can only be viewed as an attempt to circumvent the intent of our page-limit rule. *See Production and Maintenance Employees Local 504 v. Roadmaster Corp.*, 954 F.2d 1397, 1406 (7th Cir. 1992); *see also Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-89-8, 29 NRC 399, 406 n.1 (1989). We do not condone Intervenor's effort to evade our page-limits rule.

RULES OF PRACTICE: APPELLATE REVIEW; COMPLIANCE WITH COMMISSION RULES

Page limits "are intended to encourage parties to make their strongest arguments clearly and concisely, and to hold all parties to the same number of pages of argument." *Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 46. We expect parties in Commission proceedings to abide by our current page-limit rules, and if they cannot, to file a motion to enlarge the number of pages permitted.

MEMORANDUM AND ORDER

The Board of Commissioners of Orange County, North Carolina ("Orange County"), seeks Commission review of three Licensing Board decisions (LBP-00-12, LBP-00-19, and LBP-01-9) that, cumulatively, rejected Orange County's challenges to a license amendment to expand spent fuel storage capacity at the Shearon Harris nuclear power reactor in North Carolina. Orange County also seeks a stay of the final Board decision (LBP-01-9) approving the amendment. We deny the petition for review and the request for a stay.

I. BACKGROUND

This proceeding began in December 1998, when Carolina Power & Light Company ("CP&L") applied for a license amendment to increase the spent fuel storage capacity at its Shearon Harris plant. The Shearon Harris fuel handling building was originally designed and constructed with four separate storage pools to support four proposed nuclear units. Eventually, CP&L canceled three of the four Shearon Harris units, but in the meantime it had constructed all four of the storage pools. Only pools A and B, with a combined capacity of 1128 PWR fuel assemblies and 2541 BWR assemblies, are currently in service. In the license amendment at issue here, CP&L proposes to add fuel storage rack modules to spent fuel pools C and D and to place pool C in service. To activate pools C and D, CP&L must complete construction of the cooling system for the pools.

The Board granted Orange County intervenor status to challenge the application and admitted two of Orange County's technical contentions. *See* LBP-99-25, 50 NRC 25 (1999). One admitted contention dealt with criticality control measures proposed by CP&L (enrichment, burnup, and soluble boron), and the other with quality assurance steps taken by CP&L regarding the piping that had been laid up after abandonment of construction of pools C and D. *See id.* As permitted by our rules, CP&L elected to utilize the so-called "hybrid hearing procedures" set up by 10 C.F.R. Part 2, Subpart K. *See* 10 C.F.R. § 2.1109. Under the Subpart K process (10 C.F.R. §§ 2.1111-1115), the Board permitted a period for discovery, obtained the parties' written evidentiary submissions, heard oral argument, and ultimately rejected Orange County's two technical contentions on the merits. *See* LBP-00-12, 51 NRC 247, 282-83 (2000).

The Board found Orange County's criticality concerns at odds with "dispositive" regulatory history and practice, and with a recent NRC rule, 10 C.F.R. § 50.68, which "seems to contemplate the use of enrichment, burnup, and soluble boron as criticality control measures." 51 NRC at 260. As for Orange County's quality assurance-piping concerns, the Board found that CP&L and NRC Staff witnesses "with expertise in the fields of corrosion, welding, and ASME Code requirements attest . . . that the procedures that were used to substitute for construction records and examination during layup are adequate to assure a level of safety as required by the regulations." *Id.* at 278. The Board stressed that "even [Orange County's] witness" advocated "just what has been done." *Id.* The Board concluded that Orange County had presented "no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing." *Id.* at 282-83.

The Board subsequently admitted one of Orange County's environmental contentions (EC-6). *See* LBP-00-19, 52 NRC 85 (2000). Contention EC-6 posed the question whether a seven-step accident sequence, culminating in initiation of an exothermic oxidation reaction in spent fuel pools C and D, has "a probability sufficient to provide the beyond-remote-and-speculative 'trigger' that is needed to compel preparation of an EIS [environmental impact statement] relative to [the] proposed licensing action." *Id.* at 95. The seven-step sequence is as follows: (1) a degraded core accident; (2) containment failure or bypass; (3) loss of all spent fuel cooling and makeup systems; (4) extreme radiation doses precluding personnel access; (5) inability to restart any pool cooling or makeup systems due to extreme radiation doses; (6) loss of most or all pool water through evaporation; and (7) initiation of an exothermic oxidation reaction in pools C and D.

Again, pursuant to Subpart K, the Board allowed discovery, obtained written submissions from the parties, and heard oral argument.[1] On March 1, 2001,

[1] On December 21, 2000, after the Subpart K oral argument on Contention EC-6 but before issuance of the Board's merits ruling, the NRC Staff issued the license amendment. The NRC Staff made the license amendment immediately effective based on the Staff's final determination that the amendment involved no significant hazards consideration ("NSHC"). *See* 10 C.F.R. §§ 50.58(b)(5), 50.92. On December 22, Orange County petitioned for Commission review of the NSHC finding and requested a suspension and stay of the issuance of the license amendment. The Commission summarily rejected the petition, which is not permitted by our regulations. *See* CLI-01-7, 53 NRC 113, 118 (2001). Nonetheless, citing its "discretionary powers," the Commission sought additional information from the NRC Staff to determine whether "the Staff's NSHC determination requires further action by the Commission." *Id.* at 119. Further, "[t]o preserve the status quo," the Commission directed CP&L to store no spent fuel under the license amendment, pending a further order of the Commission or a Licensing Board decision approving the amendment, whichever came sooner. *See id.* The subsequent Board decision approving the Shearon Harris license amendment, which we decline to review today, renders the NSHC question inconsequential for this adjudication, and thus we do not address it further.

the Board decided Contention EC-6 on the merits. The Board ruled that: (1) the NRC Staff had met its burden to demonstrate that the accident scenario postulated by Orange County is "remote and speculative," and thus does not warrant preparation of an EIS; and (2) Orange County had failed to show a "genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy" at a further evidentiary hearing. *See* LBP-01-9, 53 NRC 239, 271 (2001). After evaluating the parties' expert submissions and probability assessments, the Board found the accident scenario's probability to lie, "conservatively," in the range of "2.0E-07 per reactor year (two occurrences in 10 million reactor years) or less," a probability estimate the Board found to be "within the category of remote and speculative matters." *Id.* at 267, 268. The Board accordingly authorized the immediate grant of CP&L's license amendment, and dismissed the proceeding. *See id.* at 271.

On March 16, 2001, Orange County petitioned for review of LBP-00-12, LBP-00-19, and LBP-01-9 and requested an emergency stay of LBP-01-9's authorization of the license grant.

II. DISCUSSION

Orange County alleges that the Board's decisions meet the Commission's standard for taking discretionary review because "they raise substantial questions with respect to their reliance on legal errors and clear factual errors. They also raise substantial and important questions of law, discretion and policy."[2] We interpret Orange County's petition as seeking review on the grounds that "[a] finding of material fact is clearly erroneous" under 10 C.F.R. § 2.786(b)(4)(i); "[a] necessary legal conclusion . . . is a departure from or contrary to established law" under 10 C.F.R. § 2.786(b)(4)(ii); and/or "[a] substantial and important question of law, policy or discretion has been raised" under 10 C.F.R. § 2.786(b)(4)(iii).[3]

We disagree with Orange County's view of the case. As we see the record, the Board fully considered Orange County's claims on the basis of extensive submissions, including Orange County's, and resolved all issues reasonably. The Board's decisions for the most part rest on its own carefully rendered fact findings, an area where we repeatedly have declined to second-guess plausible Board decisions. *See, e.g., Hydro Resources, Inc.* (P.O. Box 15910, Rio Rancho, NM 87174), CLI-01-4, 53 NRC 31, 45 (2001); *Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), CLI-98-3, 47 NRC 77, 93 (1998); *Kenneth G. Pierce* (Shorewood, Illinois), CLI-95-6, 41 NRC 381, 382 (1995).

[2] *See* "Orange County's Petition for Review of LBP-00-12, LBP-00-19, and LBP-01-09," at 7, Mar. 16, 2001 ("Orange County's Petition").

[3] Section 2.786 applies to Subpart K by virtue of 10 C.F.R. § 2.1117, which makes Subpart G rules applicable "except where inconsistent" with Subpart K. Subpart K has no rule of its own for petitions for review.

On a petition for review, Orange County must adequately call the Commission's attention to claimed errors in the Board's approach. Here, Orange County has submitted a complex set of pleadings that includes numerous detailed footnotes, attachments, and incorporations by reference. *See* Section F of this Order, *infra.* We deem waived any arguments not raised before the Board or not clearly articulated in the petition for review. *See Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 46; *Commonwealth Edison Co.* (Zion Nuclear Power Station, Units 1 and 2), CLI-99-4, 49 NRC 185, 194 (1999); *Curators of the University of Missouri*, CLI-95-1, 41 NRC 71, 132 n.81 (1995). Below, we discuss what we take to be Orange County's principal grievances, and explain why, in our judgment, they do not justify plenary Commission appellate review.[4]

A. Resolving Fact Questions in Subpart K Proceedings

We turn first to a preliminary matter that pervades Orange County's petition. The petition depends largely on the proposition that the County met its burden to justify moving forward from a Subpart K abbreviated hearing — i.e., the submission of written materials plus oral argument — to a full trial-type evidentiary hearing. According to Orange County, a factual disagreement between its expert and those of CP&L and the NRC Staff is enough to trigger a full evidentiary hearing. We think that Orange County's position oversimplifies our Subpart K process — which empowers a licensing board to *resolve* fact questions, when it can do so accurately, at the abbreviated hearing stage.

Subpart K establishes a two-part test to determine whether a full evidentiary hearing is warranted: (1) there must be a genuine and substantial dispute of fact "which can only be resolved with sufficient accuracy" by a further adjudicatory hearing; and (2) the Commission's decision "is likely to depend in whole or in part on the resolution of that dispute." *See* 10 C.F.R. § 2.1115(b). Earlier this year, we elaborated on the meaning of Subpart K by pointing to language from the Statement of Considerations for the rule. *See Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit 3), CLI-01-3, 53 NRC 22 (2001). Specifically, we stated:

> In promulgating section 2.1115(b) of Subpart K, we used the same test described in the Nuclear Waste Policy Act of 1983 ["NWPA"] at 42 U.S.C. § 10154(b)(1). We noted that
>
> > the statutory criteria are quite strict and are designed to ensure that the hearing is focused exclusively on real issues. They are similar to the standards under the Commission's existing rule for determining whether summary disposition is warranted. They go further,

[4] Safety questions not properly raised in an adjudication may nonetheless be suitable for NRC consideration under its public petitioning process, 10 C.F.R. § 2.206. *See Power Authority of the State of New York* (James A. FitzPatrick Nuclear Power Plant; Indian Point, Unit 3), CLI-00-22, 52 NRC 266, 311 (2000); *International Uranium (USA) Corp.* (Receipt of Material from Tonawanda, New York), CLI-98-23, 48 NRC 259, 265-66 (1998).

however, in requiring a finding that adjudication is necessary to resolution of the dispute and in placing the burden of demonstrating the existence of a genuine and substantial dispute of material fact on the party requesting adjudication.

See id. at 26 n.5, *quoting* Final Rule, "Hybrid Hearing Procedures for Expansion of Spent Nuclear Fuel Storage Capacity at Civilian Nuclear Power Reactors," 50 Fed. Reg. 41,662, 41,667 (Oct. 15, 1985).

Subpart K derives from the NWPA, where Congress called on the Commission to "encourage and expedite" onsite spent fuel storage. *See* 42 U.S.C. § 10151(a)(2). To help accomplish this goal, the NWPA required the Commission, "at the request of any party," to employ an abbreviated hearing process — i.e., discovery, written submissions, and oral argument. *See* 42 U.S.C. § 10154. The NWPA authorized the Commission to convene additional "adjudicatory" hearings "only" where critical fact questions could not otherwise be answered "with sufficient accuracy." *See* 42 U.S.C. § 10154(b)(1)(A). Our later-enacted Subpart K codifies in our rules the congressionally mandated abbreviated hearing process. *See* 10 C.F.R. §§ 2.1111-2.1115.

As noted in the congressional debate on the NWPA, the abbreviated hearing process was, when enacted, a "totally new procedure to be incorporated into the NRC licensing process." *See* 128 Cong. Rec. S15,644 (daily ed. Dec. 20, 1982) (statement of Sen. Mitchell). The purpose of the abbreviated hearing was "to speed up the licensing of onsite storage expansion." *See id.* The "criteria by which the Commission may decide that a full adjudicatory hearing is necessary are extremely narrow." *See id.*[5]

Orange County apparently understands Subpart K as demanding a full evidentiary hearing whenever an intervenor presents *any* material facts or expert opinion that contests positions taken by the license applicant or the NRC Staff. The County thus seemingly views Subpart K merely as an alternate form of "summary disposition." Our rules long have allowed summary disposition in cases where "there is no genuine issue as to any material fact" and where "the moving party is entitled to a decision as a matter of law." *See* 10 C.F.R. § 2.749(d); *cf.* Fed. R. Civ. P. 56 (judicial summary judgment rule). Obviously, if Orange County were correct that material fact disputes invariably require a full evidentiary hearing, there would be no real difference between our traditional summary disposition practice and Subpart K.

As a simple historical matter, however, it seems unlikely to us that Congress intended the Commission to enact Subpart K simply to replicate the NRC's existing summary disposition practice. (The Commission's summary disposition rule dates from 1972; Subpart K dates from 1985). Congress "cannot be presumed to do a futile thing." *Halverson v. Slater*, 129 F.3d 180, 184 (D.C. Cir. 1997). *Accord Independent Insurance Agents of America, Inc. v. Hawke*, 211 F.3d 638, 643 (D.C. Cir. 2000). Hence, to give real-world meaning to Subpart K's abbreviated hearing process, we construe Subpart K to extend beyond the NRC's pre-existing summary disposition practice. Unlike our summary disposition rule, which requires an additional evidentiary hearing whenever a licensing board finds, based on the papers filed, that there remains a genuine issue of material fact, Subpart K's "totally new procedure" (128 Cong. Rec. at S15,644) authorizes the board to *resolve* disputed facts based on the evidentiary record made in the abbreviated hearing, without convening a full evidentiary hearing, if the board can do so with "sufficient accuracy."

The text of Subpart K (which repeats, *verbatim*, the pertinent text of the NWPA) makes this clear. Subpart K directs the Board to "[*d*]*ispose* of any *issues* of law or *fact* not designated for resolution in an adjudicatory hearing." *See* 10 C.F.R. § 2.1115(a)(2) (emphasis added). "Issues" are, by definition, points of debate or dispute. To "dispose" of issues a board must *resolve* them. To move from Subpart K's abbreviated hearing stage to an additional evidentiary hearing, a licensing board must make a specific determination that issues "can *only* be resolved with sufficient accuracy" at such a hearing. *See* 10 C.F.R. § 2.1115(b)(1) (emphasis added).

The Statement of Considerations for Subpart K reinforces the rule's text:

> The appropriate evidentiary weight to be given an expert's technical judgment will depend, for the most part, on the expert's testimony and professional qualifications. In some circumstances, it may be possible to make such a determination without the need for an adjudicatory hearing. The presiding officer must decide, based on the sworn testimony and sworn written submissions, whether the differing technical judgment gives rise to a genuine and substantial dispute of fact that *must* be resolved in an adjudicatory hearing.

See 50 Fed. Reg. at 41,667 (1985) (emphasis added).

The short of the matter is that the NWPA and our rule implementing it (Subpart K) contemplate merits rulings by licensing boards based on the parties' written submissions and oral arguments, except where a board expressly finds that "accuracy" demands a full-scale evidentiary hearing. Subpart K's abbreviated hearing approach is in harmony with other NRC rules, such as Subparts L and M, that authorize informal adjudicatory decision-making without the panoply of full trial-type processes. *See* 10 C.F.R. § 2.1201 *et seq.* (Subpart L); 10 C.F.R. § 2.1301 *et seq.* (Subpart M).

Licensing boards are fully capable of making fair and reasonable merits decisions on technical issues after receiving written submissions and hearing oral arguments. The Commission is a technically oriented administrative agency, an orientation that is reflected in the makeup of its licensing boards. Most licensing

[5] Senator Mitchell made his comments in the context of speaking in favor of his amendment, which would have prohibited use of the abbreviated hearing process in the case of an application proposing the use of a new technology to increase onsite spent fuel storage capacity.

boards have two, and all have at least one, technically trained member. In Subpart K cases, licensing boards are expected to assess the appropriate evidentiary weight to be given competing experts' technical judgments, as reflected in their reports and affidavits. The inquiry is similar to that performed by presiding officers in materials licensing cases, where fact disputes normally are decided "on the papers," with no live evidentiary hearing. *See, e.g., Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 45; *Curators of the University of Missouri*, CLI-95-1, 41 NRC at 118-20. The NRC's administrative judges, in other words, and the Commission itself, are accustomed to resolving technical disputes without resort to in-person testimony.

There may, of course, be issues, such as those involving witness credibility, that cannot be resolved absent face-to-face observation and assessment of the witness.[6] Or there may be issues involving expert or other testimony where key questions require followup and dialogue to be answered "with sufficient accuracy." In these kinds of cases, Subpart K contemplates further evidentiary hearings. Many issues, however, particularly those involving competing technical or expert presentations, frequently are amenable to resolution by a licensing board based on its evaluation of the thoroughness, sophistication, accuracy, and persuasiveness of the parties' submissions.

The Commission does not have extensive experience with Subpart K proceedings to date. On a case-by-case basis, we generally will defer to our licensing boards' judgment on when they will benefit from hearing live testimony and from direct questioning of experts or other witnesses. If, however, a decision can be made judiciously on the basis of written submissions and oral argument, we expect our boards to follow the mandate of the NWPA and Subpart K to streamline spent fuel pool expansion proceedings by making the merits decision expeditiously, without additional evidentiary hearings. *See* 42 U.S.C. §§ 10151(a)(2), 10154.

B. Review of LBP-01-9

In LBP-01-9, 53 NRC 239, the Board addressed the question whether the seven-step severe accident sequence postulated by Orange County is remote and speculative so as not to warrant the preparation of an EIS before issuance of

[6] If, for example, the color indicated on a gauge is critical in determining the outcome of a matter, and witness A gives an affidavit stating that the gauge light was red, and witness B gives an affidavit stating that the light was green, with nothing more (such as corroborating affidavits or other documentary evidence that tends to establish the color of the gauge light), the merits of the case cannot be decided adequately or fairly based on the written submissions alone. The decision-maker must examine the live witnesses to determine, at a minimum, their demeanor, their biases, and whether they have any defects in vision. Most technical issues before NRC licensing boards fall outside this "red light/green light" category of factual disputes, which hinge on credibility of witnesses. They are more closely akin to evaluating whether the gauge was properly designed or was functioning correctly at the critical time — issues which, depending on the caliber and completeness of written submissions, may or may not necessitate hearing testimony from live witnesses.

the license amendment requested by CP&L. Orange County contends that the Board misapplied the Subpart K standard regarding going forward to a formal evidentiary hearing, that it improperly decided the merits of the dispute, and that it arbitrarily ignored or rejected Orange County's factual evidence without providing a reasoned explanation.

We disagree. As we see the case, the Board acted reasonably. It carefully described and assessed the procedures performed and assumptions made by all of the parties in answering the Board's questions regarding the probability of occurrence of the seven-step accident sequence. The Board presented a step-by-step critique of the parties' efforts, noting areas of agreement and disagreement between them, and registering its conclusions about the propriety of various assumptions made by the parties' technical witnesses. The Board's explanation of its approach was measured and persuasive.

While finding some differences in the parties' approaches up and down the accident sequence, the Board found that the cardinal points of divergence between the NRC Staff and Orange County take place at steps 4, 5, and 6. *See* LBP-01-9, 53 NRC at 258-65. At step 4 (extreme radiation levels precluding personnel access), the Board characterized Orange County's analysis as "simplistic," as it was based on the "unrealistically conservative" assumption that a fixed amount of radioactive material deposits evenly in a 200-meter circle. *Id.* at 260. The Board favored use of the Staff's more sophisticated and realistic dispersion modeling. *Id.* At step 5 (inability to restart cooling or makeup systems due to extreme radiation doses), the Board refused to accept Orange County's "unsupported surmise" that, in order to restore cooling or makeup systems, CP&L workers would be unwilling to accept 25-rem doses, which are within EPA guidelines for emergencies. *Id.* at 263. The Board deemed the NRC Staff's analysis, by contrast, "reasonably thorough and credible based on existing regulations and guidance for exposure to emergency workers." *Id.* At step 6 (loss of most or all pool water through evaporation), the Board found that Orange County, in its "assignment of certainty to this step of the sequence," had not "adequately accounted" for the "myriad ways" to get recovery makeup water into the pools. *Id.* at 265. All of the parties accepted a probability of 1.0 for step 7, initiation of an exothermic oxidation reaction in the spent fuel pools after loss of most or all of the pool water through evaporation. *See id.* at 266.

The NRC Staff, after its extensive analysis, assigned a value of 2.0×10^{-7} (once in 5 million reactor years) to the overall probability of the seven-step scenario. *See id.* After analysis by its contractor, CP&L found the probability to be even smaller — 2.7×10^{-8}.[7] *See id.* Orange County's estimate, based on the opinion of

[7] The Board viewed CP&L's analysis, enhanced by a probabilistic risk assessment, as "a beneficial, although not dispositive, confirmation of the validity of the Staff's analysis to the degree the CP&L analysis yielded a probability estimate that was equal to or lower than the Staff's estimate." *Id.* at 252.

its sole witness, Dr. Gordon Thompson, is 1.6×10^{-5}. *See id.* For the reasons given in its order and summarized above, the Board accepted the Staff's figure, labeled it conservative, and concluded that the seven-step accident scenario is remote and speculative.[8] *See id.* at 268. As we mentioned at the outset of today's decision, the Commission is generally not inclined to upset the Board's fact-driven findings and conclusions, particularly where it has weighed the affidavits or submissions of technical experts. Here, in our judgment, the Board analyzed the parties' technical submissions carefully, and made intricate and well-supported findings in a 42-page opinion. We see no basis for the Commission, on appeal, to redo the Board's work.

As we held in Section A, *supra,* the Board possessed authority under Subpart K to reach a merits decision rather than designate disputed issues of fact for resolution at a formal evidentiary hearing. *See* 10 C.F.R. § 2.1115(a)(1). None of the disputed issues, the Board found (LBP-01-9, 53 NRC at 271), could be resolved with sufficient accuracy *only* by the introduction of additional evidence at a formal hearing. *See* 10 C.F.R. § 2.1115(b)(1). This was a reasonable finding. Orange County did not challenge the qualifications of any of the Staff's or CP&L's technical witnesses. On behalf of Orange County, Dr. Thompson made suggestions regarding steps he thought should be taken to improve the analytical work done by the Staff and CP&L;[9] however, his own analysis did not take these steps. The proponent of a contention must supply, at the written submission and oral argument stages of a Subpart K proceeding, all of the facts upon which it intends to rely at the formal evidentiary hearing, should one prove necessary. *See Millstone,* CLI-01-3, 53 NRC at 27.

[8] The Commission has never determined a threshold accident probability figure for imposing the requirement of preparing an EIS. Eleven years ago, the Commission indicated that such a threshold would be "better explored outside the scope of a particular case involving only a few parties," and declined "either to endorse or reject" an Appeal Board determination that an accident probability of 10^{-4} is remote and speculative. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333, 335 (1990). In a later decision in that same proceeding, the Commission reiterated that "low probability is the key to applying NEPA's rule-of-reason test to contentions that allege that a specified accident scenario presents a significant environmental impact that must be evaluated." *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 131 (1990). Because we do not disturb the Board's finding of extremely low probability in this case, we need not decide here whether Orange County's 1.6×10^{-5} probability estimate is remote and speculative so as not to require preparation of an EIS. The Board itself similarly declined to draw a "line in the sand." *See* LBP-01-9, 53 NRC at 268.

[9] In an extensive affidavit filed in conjunction with Orange County's stay motion, Dr. Thompson advanced numerous technical criticisms of the Board's ruling in LBP-01-9. Among other things, he challenges the NRC Staff's "ARCON" methodology for modeling dispersion of radioactive materials. The ARCON model used by the Staff is conservative, takes into account site-specific meteorological conditions, and considers building wake effects to a limited degree. As in the case of all atmospheric dispersion models, the results from the ARCON model are subject to some degree of uncertainty. Despite its limitations, the ARCON model remains useful in determining whether the accident scenario at issue here is remote and speculative. The bottom line is that the Board found the NRC Staff's analysis "credible" in its own right and more persuasive than that of Dr. Thompson. *See* LBP-01-9, 53 NRC 259-60.

Notably, as the Board stressed, the NRC Staff and CP&L subjected their analytical work to peer review. *See* LBP-01-9, 53 NRC at 268-69. Orange County's expert, Dr. Thompson, did not. *See id.* at 268. The Board found that

> in the absence of any specific evidence of bias or mistake, the . . . internal review of the components of its contention EC-6 probability analysis by Staff senior technical or supervisory personnel who were not involved in preparing the Staff's analysis is adequate in this context to provide the Board with confidence in the reliability of the Staff analysis regarding all of the important issues associated with each step of the postulated sequence.

Id. at 269.

In sum, we see no basis for upsetting the Board's probability estimate or its decision against a further evidentiary hearing. Even if a further evidentiary hearing were convened, Orange County apparently intends merely to reiterate its critique of the probabilistic risk assessment of others (the NRC Staff and CP&L), but not to offer a fresh analysis of its own. *See* "Official Transcript of Proceedings" at 479-81 (Dec. 7, 2000). Under these circumstances, scheduling a further hearing would serve only to delay these proceedings and increase the costs for all parties, in direct contravention of the NWPA.

C. Review of LBP-00-19

Orange County contests the form in which Contention EC-6 was admitted. Specifically, Orange County faults the Board for limiting its inquiry to a specific seven-step accident scenario rather than focusing on the broader issue of the overall probability of a spent fuel pool accident at Shearon Harris. Orange County claims that it pleaded the broad accident probability issue with basis and specificity.

The crux of Orange County's environmental contention is that the NRC ought to have issued an environmental impact statement in connection with the license amendment requested by CP&L. *See* "Orange County's Request for Admission of Late-Filed Environmental Contentions" (Jan. 31, 1999). The Board focused on whether the specific accident proposed by Orange County in basis F.1 of the contention "has a probability sufficient to provide the beyond-remote-and-speculative 'trigger' that is needed to compel preparation of an EIS." *See* LBP-00-19, 52 NRC at 95. That accident scenario was articulated by CP&L in its contentions response. Orange County, in its contentions reply, agreed that CP&L's summary was "reasonable," but suggested rewording two phrases. *See* "Orange County's Reply to Applicant's and Staff's Oppositions to Request for Admission of Late-Filed Environmental Contentions" at 8 (Mar. 13, 2000). The Board adopted Orange County's rewording suggestions, and the contention was admitted.

At the contentions stage of this litigation, Orange County offered no specific causes for spent fuel pool accidents other than the seven-step scenario admitted by the Board. Orange County cannot now transform vague references to potential spent fuel pool catastrophes into litigable contentions. *See Duke Energy Corp.* (Oconee Nuclear Station, Units 1, 2, and 3), CLI-99-11, 49 NRC 328, 333-35 (1999) (NRC's "strict contention rule" requires "detailed pleadings"). Moreover, Orange County expressly approved the final language of its admitted environmental contention. The County should not now be heard to complain that the contention as admitted was too narrow. Therefore, we see no basis for the County's petition to review LBP-00-19.

D. Review of LBP-00-12

Orange County contends that the Board erred in LBP-00-12 by (1) ruling that the use of procedural and administrative measures for criticality control in the spent fuel pools is permissible; (2) ignoring Orange County's evidence regarding quality assurance issues; and (3) refusing to consider Orange County's argument that CP&L must seek a construction permit to use piping and equipment that was installed in the early 1980s and not used. We turn now to individual discussion of these asserted points of error.

1. Criticality Controls

Orange County alleged that criticality control measures proposed by CP&L would violate NRC regulations. Specifically, Orange County relies on General Design Criterion 62 (GDC 62), one of the General Design Criteria for Nuclear Power Plants listed in 10 C.F.R. Part 50, Appendix A. GDC 62 provides, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." *See* 10 C.F.R. Part 50, Appendix A ("General Design Criteria for Nuclear Power Plants"). Orange County maintains that the use of soluble boron and credits for fuel enrichment, burnup, and decay time limits are not "physical systems or processes," and thus violate GDC 62.

In another case we decide today, involving the Millstone spent fuel pool, we hold that the phrase "physical systems or processes" in GDC 62 does not prohibit the same administrative and procedural measures opposed by Orange County in the present case. *See Northeast Nuclear Energy Co,* (Millstone Nuclear Power Station, Unit 3), CLI-01-10, 53 NRC 353 (2001). At the Commission's invitation, Orange County and CP&L participated in *Millstone* as *amici curiae.* In view of our *Millstone* decision, nothing remains of the GDC 62 issue for further Commission review.

2. Quality Assurance Issues

Orange County contends that the Board ignored a significant portion of its evidentiary case on quality assurance issues and cites, in particular, alleged deficiencies in CP&L's video camera inspections of the piping system at issue in this license amendment application. Orange County maintains that the inspections covered only the embedded welds and not the embedded piping. Further, Orange County states that the Board assumed that the piping was inspected and failed to address evidence that only the welds were inspected.

Orange County's claim is incorrect. The Board specifically found that "all fifteen embedded welds *and their associated piping* were inspected using a high resolution camera, taking high quality pictures of everything inside the piping, longitudinal welds, circumferential welds, and piping surfaces." *See* LBP-00-12, 51 NRC at 276 (citation omitted and emphasis added). The Board pointed out that an NRC Staff expert had reviewed the videotapes from the remote camera examinations of ten of the fifteen embedded welds. *Id.* at 277. From the review and analysis of the videotapes and from available documentation, the NRC Staff "concluded that the piping and welds are conservatively designed; are several times thicker than required by ASME Code; are generally in good condition with some minor, but no major, defects; and have leaktight integrity." *Id.* The Board also stated that the steps advocated by Orange County's own expert are "just what has been done." *Id.* at 278.

On a more general plane, it hardly can be said that the Board gave short shrift to Orange County's quality assurance concerns. The Board admitted the issue for hearing, allowed discovery, obtained written evidence, and heard oral argument. The Board ultimately devoted some 11 pages of its order to discussing the quality assurance issue on the merits. *See id.* at 269-80. As we have stressed throughout today's decision, we do not ordinarily second-guess Board fact findings, particularly those reached with this degree of care. Orange County has given us no reason to do so here.

3. Construction Permit

Orange County maintains that the Board erred in refusing to consider its argument that CP&L must seek a construction permit to use the piping and equipment that were abandoned in the early 1980s. The Board ruled that the construction permit claim was not a part of Orange County's admitted contention and cannot be admitted unless it fulfills the late-filing standards set out in 10 C.F.R. § 2.714(a). *See* LBP-00-12, 51 NRC at 281. Because Orange County made no effort to address the late-filing standards, the Board precluded further consideration of the issue. *See id.* at 281-82. The Board also expressed skepticism that the amendment proposed by CP&L "is a 'material alteration' in the sense

intended by the regulations so as to require a construction permit." *See id.* at 281-82, *citing* 10 C.F.R. § 50.92(a).

We agree with the Board. Orange County was inexcusably late in attempting to introduce its construction permit claim. In addition to the claim's untimeliness, it seemingly lacks merit as a legal matter. While the term "material" is susceptible of various meanings, longstanding NRC Staff practice indicates that alterations of the type that require a construction permit are those that involve substantial changes that, in effect, transform the facility into something it previously was not or that introduce significant new issues relating to the nature and function of the facility. *See Portland General Electric Co.* (Trojan Nuclear Plant), LBP-77-69, 6 NRC 1179, 1183 (1977).[10] To trigger the need for a construction permit, the change must "essentially [render] major portions of the original safety analysis for the facility inapplicable to the modified facility." *See id.* The present case involves activation of already-built spent fuel pools, whose safety can be (and has been) adequately evaluated in the context of an ordinary license amendment. This seems to us a sensible approach.

E. Orange County's Request for Emergency Stay

In addition to seeking Commission appellate review, Orange County requested an emergency stay of LBP-01-9, pending appeal, insofar as that decision allowed the CP&L license amendment to take effect. Stays pending appellate review are governed by 10 C.F.R. § 2.788. In determining whether to grant a stay, the Commission will consider:

> (1) Whether the moving party has made a strong showing that it is likely to prevail on the merits;
> (2) Whether the party will be irreparably injured unless a stay is granted;
> (3) Whether the granting of a stay would harm other parties; and
> (4) Where the public interest lies.

See 10 C.F.R. § 2.788(e). Our decision today to deny Orange County's petition for review terminates adjudicatory proceedings before the Commission, and renders moot the County's motion for a stay pending appeal. Accordingly we deny it.

We took no action on Orange County's stay motion during our consideration of the County's petition for review because we saw no possibility of irreparable injury. The record indicates that the injury asserted by the County could not occur until at least July 2, 2001, when CP&L expects to place spent fuel pools C and D into service following testing. *See* Affidavit of R. Steven Edwards and Robert K. Kunita ¶ 11 (Mar. 29, 2001). Even after July 2, the additional spent fuel stored at Shearon Harris will total no more than 150 fuel elements in the short term (i.e., during 2001). *See id.* ¶ 15. Moreover, Orange County's claim of injury — offsite radiation exposure in the event of a spent fuel pool accident — is speculative, given the small likelihood of such an accident, and does not amount to the kind of "certain and great" harm necessary for a stay. *See Cuomo v. NRC*, 772 F.2d 972, 976 (D.C. Cir. 1985); *accord Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), ALAB-820, 22 NRC 743, 747-48 & n.20 (1985).

Of the four stay factors, "the most crucial is whether irreparable injury will be incurred by the movant absent a stay." *Alabama Power Co.* (Joseph M. Farley Nuclear Plant, Units 1 and 2), CLI-81-27, 14 NRC 795, 797 (1981). *Accord Sequoyah Fuels Corp. and General Atomics* (Gore, Oklahoma Site), CLI-94-9, 40 NRC 1, 7 (1994). Here there was (and is) no such injury.

F. Compliance with Commission Adjudicatory Rules

We close on a procedural note. The Commission's rule providing for review of decisions of a presiding officer plainly states that a "petition for review . . . must be no longer than ten (10) pages." *See* 10 C.F.R. § 2.786(b)(2). Orange County's petition for review, although nominally confined to 10 pages, resorts to the use of voluminous footnotes, references to multipage sections of earlier filings, and supplementation with affidavits that include additional substantive arguments. This can only be viewed as an attempt to circumvent the intent of our page-limit rule. *See Production and Maintenance Employees Local 504 v. Roadmaster Corp.*, 954 F.2d 1397, 1406 (7th Cir. 1992); *see also Public Service Co. of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-89-8, 29 NRC 399, 406 n.1 (1989). While we did not strike Orange County's petition, and we expanded other parties' page limits to allow them to respond fully to Orange County's submission, we do not condone the County's effort to evade our page-limits rule.

Page limits "are intended to encourage parties to make their strongest arguments clearly and concisely, and to hold all parties to the same number of pages of argument." *Hydro Resources, Inc.*, CLI-01-4, 53 NRC at 46. We are quite aware that our current 10-page limit for petitions for review (and responses) requires the parties to be direct and concise. This may be difficult in cases where, as here, the issues are numerous and complex. Hence, Orange County's effort to find creative means to avoid the page limits is in a sense understandable. Indeed, the Commission itself has invited public comment on a proposed rule that would, among other procedural reforms, increase the pages permitted for a petition for review from 10 to 25. *See* "Changes to Adjudicatory Process: Proposed Rule," 66 Fed. Reg. 19,610, 19,626 (Apr. 16, 2001).

[10] The example the Licensing Board cited in *Trojan* was a construction permit issued for alterations in the University of Maryland's research reactor. *See id.* There, the alterations involved complete removal of the existing control rods, rod drive mechanisms, core instrumentation and control room equipment and replacement of these components with components of a different design. *See id.* The dearth of other examples of post-operating license amendment construction permits supports our view that such permits are necessary only in cases of dramatic or transforming changes in existing facilities.

For now, though, we advise NRC litigants against taking Orange County's self-help approach. We expect parties in Commission proceedings to abide by our current page-limit rules, and if they cannot, to file a motion to enlarge the number of pages permitted. In the future, the Commission may exercise its authority to deal more harshly with attempts to circumvent page-limit or other procedural rules.

III. CONCLUSION

For the foregoing reasons, the Commission (1) *denies* Orange County's petition for review of the Board rulings in LBP-00-12, LBP-00-19, and LBP-01-9; and (2) *denies* Orange County's request for an emergency stay of LBP-01-9.

IT IS SO ORDERED.

For the Commission

ANDREW L. BATES for
ANNETTE L. VIETTI-COOK
Secretary of the Commission

Dated at Rockville, Maryland,
this 10th day of May 2001.

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Atomic Safety and Licensing Boards Issuances

Cite as 53 NRC 239 (2001) LBP-01-9

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Dr. Peter S. Lam
Thomas D. Murphy

In the Matter of

Docket No. 50-400-LA
(ASLBP No. 99-762-02-LA)

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant)

March 1, 2001

In this 10 C.F.R. Part 2, Subpart K spent fuel pool (SFP) expansion proceeding, in accordance with 10 C.F.R. § 2.1115, the Licensing Board denies the request of Intervenor Board of Commissioners of Orange County, North Carolina (BCOC), to designate for an evidentiary hearing an admitted BCOC contention challenging the NRC Staff's environmental assessment (EA) determination not to prepare an environmental impact statement (EIS) under the National Environmental Policy Act of 1969 (NEPA) regarding Applicant Carolina Power and Light Company's (CP&L) request to increase the spent fuel storage capacity of its Shearon Harris Nuclear Power Plant through a 10 C.F.R. § 50.90 facility operating license amendment. The Licensing Board concludes (1) there was no genuine and substantial factual or legal dispute that required resolution in an evidentiary hearing in connection with the BCOC environmental contention claiming that an EIS was needed because of the probability of SFP accidents; and (2) the NRC Staff's best estimate of probability of a BCOC-posited severe accident scenario (2.0E-7) was reasonable and demonstrated that the scenario was "remote and speculative" so as not to require preparation of an EIS.

RULES OF PRACTICE: BURDEN OF PROOF (SUBPART K PROCEEDING; ENVIRONMENTAL CONTENTIONS)

The proponent of the need for an evidentiary hearing bears the burden of establishing that need, but the Staff bears the ultimate burden to demonstrate its compliance with NEPA in its EA determination that an EIS is not necessary relative to a license amendment request. *See Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), CLI-98-3, 47 NRC 77, 89 (1998).

RULES OF PRACTICE: BURDEN OF PROOF (SUBPART K PROCEEDING; ENVIRONMENTAL CONTENTIONS)

Once an intervenor crosses the admissibility threshold relative to its environmental contention, the ultimate burden in a Subpart K proceeding then rests with the proponent of the NEPA document — the Staff (and the applicant to the degree it becomes a proponent of the Staff's EIS-related action) — to establish the validity of that determination on the question whether there is an EIS preparation trigger.

RULES OF PRACTICE: EXPERT WITNESS(ES)

When the qualifications of an expert witness are challenged, the party sponsoring the witness has the burden of demonstrating his or her expertise. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-410, 5 NRC 1398, 1405 (1977). Further, although the Federal Rules of Evidence (FRE) are not directly applicable to Commission proceedings, NRC presiding officers often look to the rules for guidance, including FRE 702 that allows a witness to be qualified as an expert " '[i]f scientific, technical, or other specialized knowledge will assist the trier of fact to understand the evidence or determine a fact in issue.' " *Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-669, 15 NRC 453, 475 (1982) (quoting FRE 702). In addition, agency caselaw indicates that the qualifications of an expert are established by showing either academic training or relevant experience, or some combination of the two. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), LBP-78-36, 8 NRC 567, 570 (1978).

NEPA: ENVIRONMENTAL ANALYSIS (REMOTE AND SPECULATIVE EVENT)

In making a determination about whether a postulated accident sequence is "remote and speculative" so as not to require an analysis of its impacts in an EIS, a Staff determination can be made without a full PRA analysis. Rather, the

Staff determination can be based on existing materials available to it, probabilistic and otherwise, supplemented by additional information it might obtain from the applicant in an environmental report or through requests for additional information (RAIs).

REGULATIONS: INTERPRETATION (10 C.F.R. §§ 20.1001(b), 20.1201(a)(1), 50.47(b)(11) IN CONJUNCTION WITH ENVIRONMENTAL PROTECTION AGENCY PROTECTIVE ACTION GUIDE 2.5)

A review of the applicable regulatory provisions — 10 C.F.R. §§ 20.1001(b), 20.1201(a)(1), 50.47(b)(11) — indicates there is no regulatory bar that prohibits the use of a 25-rem dose limit in an actual emergency or in planning a response to such an emergency to assure SFP cooling after an accident. Likewise, Environmental Protection Agency Protective Action Guide 2.5 clearly allows a dose of up to 25 rem for life saving and protection of large populations. Moreover, because this dose is within regulatory standards, the Board will not engage in the unsupported surmise that knowledgeable plant personnel would be wholly unwilling to accept such doses in an emergency such as the postulated accident sequence. Thus, it is appropriate to use a permissible dose of 25 rem for purposes of calculating stay times and probabilities that personnel at the plant would be able to perform the necessary activities required to restore SFP cooling and makeup.

RULES OF PRACTICE: EFFECT OF 10 C.F.R. § 2.1113 WRITTEN PRESENTATION IN NEPA CONTEXT

Relative to challenge to validity of Staff EA determination, information submitted by the Staff in its 10 C.F.R. § 2.1113 written presentation can establish any requisite "hard look" under NEPA has been taken. *See* 10 C.F.R. § 51.34(b).

TECHNICAL ISSUE DISCUSSED

The following technical issue is discussed: Probability of Postulated Spent Fuel Pool Accident Sequence.

MEMORANDUM AND ORDER

(Denying Request for Evidentiary Hearing and Terminating Proceeding)

Before the Licensing Board in this 10 C.F.R. Part 2, Subpart K proceeding is a challenge by Intervenor Board of Commissioners of Orange County, North

Carolina (BCOC), to a license amendment request by Applicant Carolina Power and Light Company (CP&L) that would permit CP&L to increase the spent fuel storage capacity at its Shearon Harris Nuclear Power Plant (Shearon Harris) by placing two inactive spent fuel pools (SFPs) into service. The sole remaining contention to be resolved is an environmental contention (EC) — EC-6, Environmental Impact Statement Required — that we admitted in LBP-00-19, 52 NRC 85 (2000). With this issue statement, BCOC contests the NRC Staff's December 1999 decision that the National Environmental Policy Act of 1969 (NEPA), 42 U.S.C. § 4321, and the Commission's implementing regulations, 10 C.F.R. Part 51, do not require the preparation of an environmental impact statement (EIS) relative to the CP&L SFP expansion request.

Pursuant to 10 C.F.R. § 2.1113, in December 2000 the Licensing Board entertained oral arguments by the parties concerning the pending question whether an evidentiary hearing is necessary regarding contention EC-6. BCOC asserts it has established there are genuine and substantial disputes of fact or law relative to its admitted contention that warrant an evidentiary hearing. Applicant CP&L and the Staff, however, maintain that BCOC has failed to identify any evidence of any disputed factual or legal matters that warrant an additional evidentiary proceeding, and that the Board should rule in their favor on the merits of the contention, thereby terminating this proceeding.

The Licensing Board finds that (1) BCOC has failed to show there is a genuine and substantial dispute of fact or law that only can be satisfactorily resolved by a further evidentiary hearing; and (2) based on the record before us, the Staff has met its burden in demonstrating that the accident scenario postulated by BCOC in support of contention EC-6 is remote and speculative so as not to warrant the preparation of an EIS in connection with the CP&L SFP amendment request. Further, because all matters before the Board in connection with the requested amendment have been resolved in favor of amendment issuance without the need for further evidentiary presentations, we authorize the grant of the requested license amendment, effective immediately, and dismiss this proceeding.

I. BACKGROUND

The portion of this litigation currently before the Board has its basis in a December 23, 1998 CP&L application for a 10 C.F.R. § 50.90 facility operating license amendment to increase the spent fuel storage capacity at its Shearon Harris facility by adding rack modules to previously inactive SFPs C and D and place those pools into service. Responding to the application, in January 1999 the Staff published a notice of proposed no significant hazards consideration and opportunity for a hearing regarding the CP&L application. *See* 64 Fed. Reg. 2237 (Jan. 13, 1999). Subsequently, in February 1999 BCOC filed a request for hearing and petition to intervene, which it followed with a contentions supplement petition in April 1999. *See* [BCOC] Request for Hearing and Petition to Intervene (Feb. 12, 1999); [BCOC] Supplemental Petition to Intervene (Apr. 5, 1999). In addition to putting forth three contentions that raised technical concerns regarding the proposed SFP expansion, in issue statements labeled EC-1 through EC-5, BCOC claimed that CP&L and the Staff had failed to comply with various NEPA requirements, as implemented by the agency in 10 C.F.R. Part 51. In our July 1999 ruling on the BCOC intervention petition, in addition to finding admissible two BCOC technical contentions claiming the CP&L expansion measure involved inadequate criticality prevention and quality assurance measures, the Board also noted that the Staff had decided to issue an environmental assessment (EA) regarding the CP&L application and dismissed the BCOC NEPA contentions, albeit without prejudice to those matters being raised once the Staff's EA was done. *See* LBP-99-25, 50 NRC 25, 38-39 (1999).

Thereafter, pursuant to 10 C.F.R. § 2.1109, CP&L timely invoked the hybrid hearing procedures articulated in Subpart K of Part 2 relative to the further litigation of admitted contentions in this proceeding. In accordance with those procedures, after a discovery period and receipt of the parties' 10 C.F.R. § 2.1113 written summaries detailing all the known facts, data, and arguments to support or refute the existence of a genuine and substantial factual dispute, in January 2000 the Board heard oral argument on the question whether a dispute existed such that an evidentiary hearing would be necessary for all or a part of the admitted BCOC technical contentions. Ultimately, in a May 2000 decision, the Board concluded such a dispute did not exist and that CP&L had met its burden of showing that, relative to BCOC's concerns, CP&L's proposed spent nuclear fuel storage expansion was in compliance with applicable statutory and regulatory requirements. *See* LBP-00-12, 51 NRC 247, 282-83, *petition for review denied as premature,* CLI-00-11, 51 NRC 297 (2000).

In so ruling, we noted that our determination did not terminate this proceeding because certain environmental issues remained outstanding. *See id.* at 282 n.14. In this regard, on December 15, 1999, the Staff issued an EA with a finding of no significant impact (FONSI) for the proposed CP&L license amendment for Shearon Harris. *See* 64 Fed. Reg. 71,514, 71,516 (Dec. 21, 1999). In response to this Staff determination that no EIS was required, on January 31, 2000, BCOC filed a request for the admission of four late-filed environmental contentions, numbered EC-1 through EC-4, the admissibility of which were contested by CP&L and the Staff. In an August 7, 2000 ruling, the Board found that the first of these contentions, which we renumbered EC-6, was admissible. *See* LBP-00-19, 52 NRC at 93-98. This BCOC contention states:

> In the Environmental Assessment ("EA") for CP&L's December 23, 1998, license amendment application, the NRC Staff concludes that the proposed expansion of spent fuel

storage capacity at the Shearon Harris nuclear power plant will not have a significant effect on the quality of the human environment. Environmental Assessment and Finding of No Significant Impact Related to Expanding the Spent Fuel Pool Stage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 10 (December 15, [1999]). Therefore, the Staff has decided not to prepare an Environmental Impact Statement ("EIS") for the proposed license amendment. The Staff's decision not to prepare an EIS violates the National Environmental Policy Act ("NEPA") and NRC's implementing regulations, because the Finding of No Significant Impact ("FONSI") is erroneous and arbitrary and capricious. In fact, the proposed expansion of spent fuel pool storage capacity at Harris would create accident risks that are significantly in excess of the risks identified in the EA, and significantly in excess of accident risks previously evaluated by the NRC Staff in the EIS for the Harris operating license. These accident risks would significantly affect the quality of the human environment, and therefore must be addressed in an EIS.

There are two respects in which the proposed license amendment would significantly increase the risk of an accident at Harris:

(1) CP&L proposes several substantial changes in the physical characteristics and mode of operation of the Harris plant. The effects of these changes on the accident risk posed by the Harris plant have not been accounted for in the Staff's EA. The changes would significantly increase, above present levels, the probability and consequences of potential accidents at the Harris plant.

(2) During the period since the publication in 1979 of NUREG-0575, the NRC's Generic Environmental Impact Statement ("GEIS") on spent fuel storage, new information has become available regarding the risks of storing spent fuel in pools. This information shows that the proposed license amendment would significantly increase the probability and consequences of potential accidents at the Harris plant, above the levels indicated in the GEIS, the 1983 EIS for the Harris operating license, and the EA. The new information is not addressed in the EA or the 1983 EIS for the Harris operating license.

Accordingly, the Staff must prepare an EIS that fully considers the environmental impacts of the proposed license amendment, including its effect on the probability and consequences of accidents at the Harris plant. As required by NEPA and Commission policy, the EIS should also examine the costs and benefits of the proposed action in comparison to various alternatives, including Severe Accident Mitigation Design Alternatives ("SAMDAs") and the alternative of dry storage.

See id. at 93-94 (footnote omitted).

As we noted in our decision admitting this contention, all the parties agreed that the standard mandating EIS preparation is whether the action at issue is a major federal action having a significant impact on the human environment. Furthermore, the parties agreed that the agency in an EIS is not required to address consequences of an action that are remote and speculative. *See id.* at 94-95. In the context of this contention, however, the parties disagreed as to what constitutes a remote and speculative event. In its argument, BCOC identified a scenario that, as summarized by CP&L with modifications by BCOC, consisted of the following seven-step chain of events:

(1) a degraded core accident;
(2) containment failure or bypass;
(3) loss of all spent fuel cooling and makeup systems;
(4) extreme radiation doses precluding personnel access;
(5) inability to restart any pool cooling or makeup systems due to extreme radiation doses;
(6) loss of most or all pool water through evaporation; and
(7) initiation of an exothermic oxidation reaction in pools C and D.

Id. at 95. Noting the Commission's guidance on admission of such a NEPA-related issue statement in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990), and *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129 (1990), the Board admitted contention EC-6 because the materials presented by BCOC, including a 1993 individual plant examination (IPE) of core damage frequency (CDF) for the Shearon Harris facility, were "sufficient to establish a genuine material dispute of fact or law adequate to warrant further inquiry relative to the other aspects of the BCOC scenario and the associated probability analysis." LBP-00-19, 52 NRC at 97-98 (footnote omitted). In addition, the Board requested that the parties address the following three questions so that the Board could more accurately evaluate the materials provided by their section 2.1113 written summaries:

1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on p. 95, *supra*? The estimates should utilize plant-specific data where available and should utilize the best available generic data where generic data are relied upon.

2. The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the [Advisory Committee on Reactor Safeguard's (ACRS)] April 13, 2000 letter suggest that the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an exothermic reaction, greater than was assumed in the decade-old NUREG-1353)?

3. Assuming the Board should decide that the probability involved is of sufficient moment so as not to permit the postulated accident sequence to be classified as "remote and speculative," what would be the overall scope of the environmental impact analysis the Staff would be required to prepare (i.e., limited to the impacts of that accident sequence or a full blown EIS regarding the amendment request)?

Id. at 98-99.

Following a 2-month discovery period,[1] on November 20, 2000, the parties filed their section 2.1113(a) summaries, with accompanying witness affidavits and documentary exhibits, in support of their respective positions on whether there is a genuine and substantial factual dispute that requires resolution in an evidentiary hearing as well as the efficacy of the Staff's EA determination that an EIS is not required for the CP&L amendment. *See* Summary of Facts, Data, and Arguments on Which Applicant Proposes to Rely at the Subpart K Oral Argument Regarding Contention EC-6 (Nov. 20, 2000) [hereinafter CP&L Summary]; NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon Which the Staff Proposes to Rely at Oral Argument on Environmental Contention EC-6 (Nov. 20, 2000) [hereinafter Staff Summary]; Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with Respect to the Need to Prepare an Environmental Impact Statement to Address the Increased Risk of a Spent Fuel Pool Accident (Contention EC-6) (Nov. 20, 2000) [hereinafter BCOC Summary]. And in support of their summary statements, each of the parties took a somewhat different approach. BCOC places particular reliance on its supporting witness, Dr. Gordon Thompson, and a November 2000 report he prepared giving his views on the probability of a release from the Shearon Harris SFPs as a result of a severe reactor accident. *See* BCOC Summary, Exh. 2 (Dr. Gordon Thompson, The Potential for a Large, Atmospheric Release of Radioactive Material from Spent Fuel Pools at the Harris Nuclear Power Plant: The Case of a Pool Release Initiated by a Severe Reactor Accident (Nov. 20, 2000)) [hereinafter Thompson Report]. CP&L, on the other hand, provided what it claims is a "state-of-the-technology" probabilistic analysis, the so-called ERIN report, done by a contractor specifically to address the BCOC contention. *See* CP&L Summary, Exh. 1, attach. C (ERIN Engineering and Research, Inc., Technical Input for Use in the Matter of Shearon Harris Spent Fuel Pool Before the Atomic Safety and Licensing Board (Nov. 2000)) [hereinafter ERIN Report]. For its part, as outlined in the affidavit of the four Staff witnesses that accompanied the Staff's written summary, *see* Affidavit of Gareth W. Parry, Stephen F. LaVie, Robert L. Palla, and Christopher Gatton in Support of NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon Which the Staff Proposes to Rely at Oral Argument on [EC-6] (Nov. 17, 2000) [hereinafter Staff Affidavit], the Staff addresses the contention by providing an analysis of existing CP&L probabilistic risk assessment (PRA)-related documents, principally an August 1993 IPE; a June 1995 individual plant examination for external events (IPEEE); and a 1995 probabilistic safety study (PSA) that updates the 1993 IPE, and other existing information relating to the Shearon Harris facility, including NUREG-1488, Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plants East of the Rocky Mountains (Apr. 1994), a seismic hazards analysis for sixty-nine nuclear power plants east of the Rocky Mountains; SECY-00-0007, Proposed Staff Plan for Low Power and Shutdown Risk Analysis Research to Support Risk-Informed Regulatory Decision-Making (Jan. 12, 2000), concerning low power or shutdown degraded core probabilities; portions of the Shearon Harris facility Final Safety Analysis Report (FSAR); CP&L information submitted in support of its December 1998 application; information obtained in discovery; and a September 2000 facility tour, *see* Staff Summary at 27-35. Subsequently, on December 7, 2000, the Licensing Board held a day-long oral argument with respect to contention EC-6 in Raleigh, North Carolina.[2] *See* Tr. at 443-706.

II. ANALYSIS

A. Standards Governing 10 C.F.R. § 2.1115 Determination Regarding the Need for an Evidentiary Hearing to Resolve Admitted Issues

1. The Nuclear Waste Policy Act of 1982 and Implementing Regulations

The procedures in 10 C.F.R. Part 2, Subpart K were established in response to a congressional mandate found in the Nuclear Waste Policy Act of 1982 (NWPA). NWPA § 134, 42 U.S.C. § 10154, states:

> (a) Oral Argument.—In any Commission hearing under section 189 of the Atomic Energy Act of 1954 (42 USC 2239) on an application for a license, or for an amendment to an existing license . . . to expand the spent nuclear fuel storage capacity at the site of a civilian nuclear power reactor . . . the Commission shall . . . provide an opportunity for oral argument . . . . The oral arguments shall [be] preceded by such discovery procedures as the rules of the Commission shall provide. The Commission shall require each party . . . to submit in written

[1] In accord with the 60-day schedule established by the Board in LBP-00-19, 52 NRC at 100, the formal discovery period relative to this BCOC contention began on August 21, 2000, and was scheduled to conclude on October 20, 2000. On October 13, 2000, BCOC filed a motion for an extension of time for discovery, briefing, and oral argument, requesting that the Board extend the discovery period to the full 90-day period permitted under 10 C.F.R. § 2.1111. *See* [BCOC] Motion for Extension of Schedule for Discovery, Briefing and Oral Argument and Request for Expedited Consideration at 2-9 (Oct. 13, 2000). The Board denied the BCOC request on the basis, among other things, that the requested extension was not justified under that provision's "exceptional circumstances" standard. *See* Licensing Board Memorandum and Order (Denying Discovery Deadline Extension Motion) (Oct. 19, 2000) at 3-4 (unpublished).

[2] On December 21, 2000, the Staff notified the Board and the other parties that, in accordance with 10 C.F.R. § 50.91, on that date it had issued a final no significant hazards consideration determination and a license amendment authorizing the requested SFP expansion at the Shearon Harris facility. *See* Board Notification 2000-06 (Dec. 21, 2000). By memorandum and order dated February 14, 2001, the Commission directed CP&L not to store spent fuel under the license amendment pending further Commission order or a Board order approving the amendment. *See* CLI-01-7, 53 NRC 113, 119 (2001).

form . . . a summary of the facts, data, and arguments upon which such party proposes to rely . . . .

(b) Adjudicatory Hearing.—(1) At the conclusion of any oral argument under subsection (a), the Commission shall designate any disputed questions of fact, together with any remaining questions of law, for resolution in an adjudicatory hearing only if it determines that—

(A) there is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing; and

(B) the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute.

Sections 2.1113 and 2.1115 of Title 10 of the *Code of Federal Regulations* incorporate these requirements as mandated by the NWPA. Thus, section 2.1115(a)(1), (2) provides that

[a]fter due consideration of the oral presentation and the written facts and data submitted by the parties and relied on at the oral argument, the presiding officer shall promptly by written order:

(1) Designate any disputed issues of fact, together with any remaining issues of law, for resolution in an adjudicatory hearing; and

(2) Dispose of any issues of law or fact not designated for resolution in an adjudicatory hearing.

Moreover, a two-part test for determining whether an evidentiary hearing is required for resolution of the issues is articulated in section 2.1115(b):

(1) There is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing; and

(2) The decision of the Commission is likely to depend in whole or in part on the resolution of that dispute.

2. *Burden of Proof*

Also relevant to our determination here is the question of the burden of proof. In this Subpart K proceeding, the parties disagree as to who bears the ultimate burden of proof regarding the merits of the BCOC environmental contention. For its part, BCOC argues that, as the Board indicated in LBP-00-12, 51 NRC at 254-55, with respect to technical contentions, although the burden of proof for demonstrating the existence of a genuine and substantial factual dispute so as to require an evidentiary hearing is on the party seeking that hearing, the ultimate burden to demonstrate that an EIS is unnecessary belongs to the Staff and the Applicant. *See* BCOC Summary at 14-15 (citing *Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), LBP-96-25, 44 NRC 331, 338 (1996)); Tr. at 461-63, 673-76. CP&L and the Staff disagree with this assessment. Citing a Licensing Board decision in *Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), LBP-96-2, 43 NRC 61, 90, *rehearing granted in part and denied in*

part, CLI-96-7, 43 NRC 235 (1996), and judicial holdings in *Citizen Advocates for Responsible Expansion, Inc. v. Dole,* 770 F.2d 423 (5th Cir.), *rehearing en banc denied,* 777 F.2d 701 (5th Cir. 1985), and *Louisiana v. Lee,* 758 F.2d 1081 (5th Cir. 1985), *cert. denied,* 475 U.S. 1044 (1986), they declare that although BCOC does not bear the ultimate burden of proof regarding the propriety of the Staff's EA determination that an EIS is not necessary, BCOC still has the burden of showing there is an accident sequence that goes beyond the "remote and speculative" threshold so as to require that the Staff then shoulder that ultimate burden by, for instance, establishing that the accident sequence does not have to be considered anyway or is not going to have any significant impacts other than those already discussed in its EA analysis. *See* CP&L Summary at 17; Staff Summary at 8-9, 36-37; Tr. at 647-48, 666-72.

We agree with BCOC that as the proponent of the need for an evidentiary hearing it bears the burden of establishing that need, but that the Staff bears the ultimate burden to demonstrate its compliance with NEPA in its determination that an EIS was not necessary relative to the CP&L SFP expansion request. *See Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), CLI-98-3, 47 NRC 77, 89 (1998). As we understand it, the crux of the argument by CP&L and the Staff is that, despite having provided a litigable contention in connection with the question of whether there is a non-remote and speculative accident sequence that requires EIS consideration, in the context of this Subpart K proceeding BCOC still has the burden of establishing that the accident sequence it has posited is indeed not remote and speculative. We do not agree. Once BCOC crossed the admissibility threshold relative to its accident sequence contention, the ultimate burden in this Subpart K proceeding then rested with the proponent of the NEPA document — the Staff (and the Applicant to the degree it becomes a proponent of the Staff's EIS-related action) — to establish the validity of that determination on the question whether the accident sequence is an EIS-preparation trigger.[3]

B. "Expert" Status of BCOC Witness Dr. Gordon Thompson

Also in controversy are the "expert" qualifications of BCOC's sole supporting affiant, Dr. Gordon Thompson. As previously noted, BCOC has proffered Dr. Thompson as an expert on nuclear power plant design and operation and provided a November 2000 report prepared by Dr. Thompson as one of the principal

[3] Although it might be asserted that the section 2.1115(b) burden imposed on BCOC as the party seeking an evidentiary hearing to establish there are appropriate factual or legal disputes is the equivalent of the "burden to go forward" that is normally ascribed to an intervenor challenging a license application, *see Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-262, 1 NRC 163, 191 (1975), this does not account for the fact that an intervenor generally is accorded the opportunity to build its case on the basis of witness cross-examination alone, *see Tennessee Valley Authority* (Hartsville Nuclear Plant, Units 1A, 2A, 1B, and 2B), ALAB-463, 7 NRC 341, 356 (1978). Nor does this assertion account for the post-Subpart K revision to the 10 C.F.R. § 2.714 standards for the admission of contentions that enhanced the showing needed for litigable issue statements.

supporting sources for its claims about the need for an evidentiary hearing. *See* BCOC Summary at 15-21; Tr. at 511-14, 518-20, 684-85. Both CP&L and the Staff, however, contest Dr. Thompson's expertise relative to the matters at issue in this proceeding.[4] *See* CP&L Summary at 20-28; Staff Summary at 18-23; Tr. at 535-37, 650-51, 702.

When the qualifications of an expert witness are challenged, the party sponsoring the witness has the burden of demonstrating his or her expertise. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-410, 5 NRC 1398, 1405 (1977). Further, although the Federal Rules of Evidence (FRE) are not directly applicable to Commission proceedings, NRC presiding officers often look to the rules for guidance, including FRE 702 that allows a witness to be qualified as an expert " '[i]f scientific, technical, or other specialized knowledge will assist the trier of fact to understand the evidence or determine a fact in issue.' " *Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-669, 15 NRC 453, 475 (1982) (quoting FRE 702). In addition, agency caselaw indicates that the qualifications of an expert are established by showing either academic training or relevant experience, or some combination of the two. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), LBP-78-36, 8 NRC 567, 570 (1978).

In the first phase of this proceeding, which addressed the two admitted BCOC-proffered technical contentions, the Staff argued that Dr. Thompson did not qualify as an expert witness based on his knowledge, skill, experience, training, or education. The Staff maintained that Dr. Thompson was not qualified to render an expert opinion on spent fuel criticality and hence argued that his opinion testimony related to the contention at issue, TC-2, should be disregarded. Noting the Staff's objection to his testimony, the Board refrained from making a bench ruling declaring him ineligible to provide expert testimony, but later held that by reason of his experience and training, "his expertise relative to reactor technical issues seems largely policy-oriented rather than operational." LBP-00-12, 51 NRC at 267 n.9. The Board also noted that it would give his testimony "appropriate weight commensurate with his expertise and qualifications" regarding issues of criticality prevention. *Id.*

In the present phase of this proceeding, BCOC reaffirms the expert qualifications of Dr. Thompson, and argues that the Board should re-evaluate its finding in LBP-00-12 that Dr. Thompson's opinions were largely "policy oriented" in that: (1) the Board overlooked his extensive knowledge relating to nuclear power plant operation and design; and (2) the contention now at hand involves new technical topics — probabilistic risk assessment and the phenomenology of spent fuel storage — that were not addressed in the previous phase of this proceeding. *See* BCOC Summary at 16. In support of the former assertion, BCOC delineates Dr. Thompson's various qualifications relating to those subjects.

According to BCOC, Dr. Thompson is highly qualified to give expert testimony relative to contention EC-6 based on his education, training, and experience. BCOC points out that Dr. Thompson received a bachelor's degree in mechanical engineering, mathematics, and physics from the University of New South Wales and later received a doctoral degree from Oxford University in the area of applied mathematics. *See id.; see also id.* Exh. 1, at 2-4 (Nov. 20, 2000 declaration of Dr. Thompson), attach. A (Gordon Thompson curriculum vitae). BCOC stresses that Dr. Thompson has more than 20 years of experience relating to nuclear facilities and their associated risks, noting that, in addition to the year he has had becoming intimately familiar with the Shearon Harris plant, Dr. Thompson also evaluated design and accident risk considerations for an array of nuclear facilities around the world. And of particular importance to this proceeding, BCOC declares, is his familiarity with probabilistic risk assessments (PRAs), including both general studies using PRA analysis and a number of studies regarding accident risks posed by plant operations and SFP storage. *See id.* at 17-21.

While Dr. Thompson may have little experience in the actual operation of a nuclear power plant or in PRA preparation, *see* CP&L Summary, Exh. 8, at 9-15, 17-20 (Oct. 16, 2000 deposition of Gordon R. Thompson), given his education and experience relating to nuclear facility and SFP design, particularly his experience with spent fuel storage issues and his previous activities with probability assessments, we cannot say that his testimony will not aid the Board in determining and/or understanding the probability of the seven-step accident sequence. Therefore, we give Dr. Thompson's testimony due weight in the subject areas in which we believe he possesses knowledge and experience that can aid the Board in its determinations regarding EC-6.

With these items resolved, we turn to the BCOC contention at issue.

C. Contention EC-6 — Accident Scenario Probability

As admitted, BCOC's contention EC-6 challenges the NRC Staff's EA determination not to prepare an EIS on the ground that the proposed CP&L license amendment is a major federal action having a significant impact on the human environment because the seven-event accident scenario identified by BCOC is not remote and speculative. In our determination admitting this contention, the Board included an extensive discussion of the Appeal Board and Commission decisions in the decade-old *Vermont Yankee* SFP expansion proceeding in which a similar NEPA concern was raised. *See* LBP-00-19, 52 NRC at 95-97. There, the Commission concluded that "future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual

[4] BCOC has not challenged the qualifications of the witnesses proffered by CP&L or the Staff in support of their written summaries. Our review of their qualifications provides us with no reason to do so either.

probabilities and accident scenarios being analyzed." *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 132 (1990). Further, the Commission indicated that although a finding that the probability for an entire accident sequence was 1×10^{-4} per reactor year (i.e., 1E-04 per reactor year in scientific notation, or one occurrence in 10,000 reactor years) should be returned to the Commission for further consideration, a lower probability would be subject to the presiding officer's judgment regarding the remote and speculative nature of the accident. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333, 336 (1990).

Consistent with this guidance and the first question we posed in LBP-00-19, each of the parties addressed the seven-item BCOC-postulated accident sequence in terms of the probabilities involved at each step (or at related steps) and for the sequence as a whole. We, in turn, address the parties' approach to each step of the postulated scenario in seeking to determine whether there are factual or legal disputes that warrant further exploration in an evidentiary hearing and, if not, whether the probability assigned to the entire scenario falls into the category of "remote and speculative" so as not to require further NEPA analysis.

In doing so, however, we provide one general observation regarding the methodology utilized by CP&L, which consisted essentially of preparing a new PRA for the contention EC-6 accident scenario, as contrasted with the analytical efforts of the Staff and BCOC. In posing the first question, we did not ask, nor did we expect, that the parties would undertake an entirely new PRA for this contention. Indeed, to do so would suggest, incorrectly in our view, that Staff EA determinations on issues like that raised in contention EC-6 cannot be made without a full PRA analysis. Instead, our request for a best estimate was intended to obtain the fruits of the type of analysis that we anticipate the Staff generally would undertake in reaching such a determination, i.e., one based on existing materials available to it, probabilistic and otherwise, supplemented by additional information it might obtain from the Applicant in an environmental report or through requests for additional information (RAIs). As it turns out, the analysis undertaken by the Staff did indeed most closely follow the process that we anticipated would be utilized to answer the first question. Thus, as between CP&L and the Staff, the Staff's analysis is the one to which we have looked in the first instance relative to BCOC's competing claims regarding the probabilities involved in the different steps of the contention EC-6 accident sequence, while viewing CP&L's PRA-enhanced analysis as a beneficial, although not dispositive, confirmation of the validity of the Staff's analysis to the degree the CP&L analysis yielded a probability estimate that was equal to or lower than the Staff's estimate.

1. *Event 1 — A Degraded Core Accident*

This first step in BCOC's postulated sequence of events leading to an exothermic reaction in the SFP assumes a serious reactor accident in which the core becomes damaged to the degree that radioactive material normally contained within the fuel rods in the core is released into the reactor and subsequently into the reactor containment building. *See* Thompson Report at 24-26; CP&L Summary at 56; Staff Summary at 27-30; Tr. at 467-72, 539-41.

a. *BCOC Position*

In its discussion regarding event one of the contention EC-6 scenario, BCOC relies on the November 20, 2000 declaration of BCOC's sole witness, Dr. Gordon Thompson and his November 2000 report, *see* Thompson Report at 24-26, 48; *id.*, App. C (Level 1 PRA analysis). And relative to this part of the scenario, BCOC references CP&L's 1993 IPE, its 1995 IPEEE, and its 1995 PSA analyses as the basis for its estimate of the degraded core accident sequence probability as the starting point of the overall sequence. BCOC actually describes four degraded core sequences that have as common features a loss of high-pressure coolant injection, loss of feedwater to the steam generators, and failure of reactor coolant pump seals, all of which lead finally to a loss of cooling to the fuel pools. BCOC also asserts that, instead of relying on the seismic hazard curves developed by the Electric Power Research Institute (EPRI) that were used by CP&L in its PSA seismic component, BCOC adjusted the core accident estimated probability to reflect Staff-endorsed seismic hazard curves from Lawrence Livermore National Laboratory. *See* Thompson Report at 25-26; *id.*, App. C at C-2 to -3. BCOC's estimate for the degraded core accident portion of the overall sequence thus is 3.1E-05 per reactor year. *See* Thompson Report at 48 (Table 1, Estimated Probability of a Degraded-Core Accident at Harris, Selected Sequences).

b. *CP&L Position*

CP&L did not calculate a specific probability for event one with respect to a degraded core accident or, as CP&L and the Staff refer to it, the CDF. Instead, CP&L evaluated event one and event two — containment failure or bypass — using PRA techniques. CP&L's analysis included internal events as initiators, such as steam generator tube rupture, loss of coolant accident, or station blackout. In addition, CP&L used the 1995 Harris plant IPEEE for determining probabilities from external events, such as fires and seismic events, and shutdown events. The results of these estimate analyses were presented in the ERIN Report and summarized in an affidavit by CP&L's expert Dr. Edward T. Burns. *See* ERIN Report at 4-1 to -76; CP&L Summary, Exh. 1, at 9-10 (Affidavit of

Edward T. Burns, Ph.D.) [hereinafter Burns Affidavit]. The CP&L best estimate of the combined probability for events one and two of the postulated sequence is summarized in the ERIN Report at Table 5-1 and was determined to be 7.67E-06. *See* Burns Affidavit at 14.

c. Staff Position

The Staff determined by analysis of existing CP&L and Staff reports that the best estimate of CDF probability at the Harris plant, including contributions from internal and external initiating events from full power, low power, and shutdown states, is 1.2E-04. *See* Staff Summary at 28, Staff Affidavit at 15-20. The Staff claims its determination of the CDF is likely to be conservative, since the frequency of initiating events has been shown to be considerably lower than assumed in the 1993 IPE, a principal document used by the Staff in its determination of the CDF. *See* Staff Affidavit at 117; *compare id.*, Exh. 9, § 3, at 45 ([CP&L] Probabilistic Safety Assessment (Rev. 1 Oct. 1995) (Table 3-17)) *with id.*, Exh. 6, at 3-18 ([CP&L] Individual Plant Examination Submittal (Aug. 1993) (Table 3-4)).

d. Board Analysis

The record before us makes it apparent that, by any measure, the degraded core accident that is the first step of BCOC's postulated sequence is a low frequency occurrence. The Staff estimates a CDF of 1.2E-04 per reactor year and BCOC puts a CDF at 3.1E-05 per reactor year. Although an argument can be made that BCOC's lower number utilizes appropriate conservatisms, nonetheless we accept the Staff's higher probability number as an appropriate starting point for the sequence. Moreover, in light of our adoption of the Staff's number, nothing regarding BCOC's assertions in connection with this aspect of the overall sequence evidences a dispute that warrants an evidentiary hearing.

2. Event 2 — Containment Failure or Bypass

The second step of BCOC's postulated sequence assumes that the reactor containment building is breached such that radioactive material within the reactor building or radioactive material within the reactor coolant system bypasses the reactor containment and is disbursed in other plant buildings or in the environment outside the reactor containment building. *See* Thompson Report at 26-29; CP&L Summary at 56; Staff Summary at 30-31; Tr. at 472-75, 542-45, 576-83, 627.

a. BCOC Position

BCOC draws on information from NUREG-1570, Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture (Mar. 1998), *see* Thompson Report, App. D, at 1 (Level 2 PRA analysis), to calculate the probabilities of a containment bypass. Although noting that other modes of containment failure may exist, BCOC discusses only the estimated probability of temperature-induced steam generator tube rupture (TI-SGTR) for selected degraded core accident sequences at Shearon Harris in its analysis of the containment failure/bypass step of the postulated sequence. Without including other mechanisms of containment failure, BCOC estimates the probability of containment bypass at 0.5 (50%). *See* Thompson Report at 26-28, *id.* at 49 (Table 2, Estimated Probability of [TI-SGTR] for Selected Degraded-Core Accident Sequences at Harris).

b. CP&L Position

See section II.C.1.b above.

c. Staff Position

The Staff again relies on CP&L's IPE and PSA as the starting point for its analysis of the probability of containment failure and/or bypass, as supplemented by Staff reports and responses to Staff discovery. *See* Staff Affidavit at 28-30. The Staff used a conditional containment failure probability of 0.2 (20%) for its analysis relative to the seven-step sequence. *See id.* at 31-32. In deriving this figure, the Staff provides an analysis of various failure modes, including early containment failures, *see id.* at 32-37; late containment failures, *id.* at 37-40; very late containment failures, *id.* at 40-41; containment isolation failures, *id.* at 41-42; and containment bypass failures, *id.* at 42-46. In this regard, the Staff assigns a conditional failure probability for a TI-SGTR of 0.021 (2.1%). *See id.* at 45. Furthermore, the Staff determined that a probability of 0.1 (10%) should be assigned to those containment failures of most concern, namely early and late containment failures. *See id.* at 69-71.

d. Board Analysis

The Board views BCOC's analysis as too simplistic for several reasons. BCOC concentrates its overall containment failure or bypass argument on the probability of a TI-SGTR and without giving adequate consideration to the specific details of accident scenarios, containment and equipment configuration, and plant operating procedures that will affect the overall probability for containment failure or bypass. In this regard, BCOC has not considered, for example, the recent

procedural changes adopted by CP&L not to run reactor coolant pumps after a severe accident. *See* Tr. at 543-45. Nor does BCOC consider the timing of containment failure based on various accident scenarios and has not linked various containment failure or bypass modes with specific core damage scenarios. In contrast, the Staff provides a credible analysis of the various containment failure and bypass modes that could be experienced at the Harris plant that is sufficient, in our estimation, to establish the validity of its estimate without the need for a further evidentiary hearing on this portion of the postulated accident scenario.[5]

In any event, regardless of these analytical differences, BCOC and the Staff do not differ significantly in their analyses of the cumulative probabilities of the postulated sequence through event two. As shown in the table below, *see infra* p. 267, at step two each of the parties shows a probability on the order of 1E-05 per reactor year. The need to utilize further evidentiary proceedings relative to this sequence step thus is not evident. Further, because the parties appear to agree on the overall probability of the basic mechanisms of accident damage and the immediate consequences of those damage mechanisms, our conclusion regarding the sufficiency of the Staff's EA determination relative to BCOC's EC-6 concern is based principally on our review of the parties' analyses of the remaining events.

3. *Event 3 — Loss of Spent Fuel Pool Cooling*

This step in BCOC's postulated sequence assumes that, as a result of the two accident sequence events discussed above, the ability to cool or provide makeup water to cover the highly radioactive spent fuel stored in the SFPs is lost. *See* Thompson Report at 29; CP&L Summary at 57; Staff Summary at 28-30; Tr. at 481-84, 545-47.

a. *BCOC Position*

BCOC asserts that for the selected accident sequences it utilized for event one, it is a certainty, i.e., a probability of 1.0 (100%), that the spent fuel system would become inoperative due to either failure of electric power on the site,

[5] In the context of this event and related event four concerning access preclusion by high radiation levels, BCOC also raises concerns about factual disputes relating to the radiological effects of high burnup fuel in the event of a containment breach or bypass, building wake effects relative to radioactive dispersion, and the Staff's use of the ARCON computer dispersion model. *See* Thompson Report at 28-29; Tr. at 475-77. Relative to high burnup fuel, in addition to the lack of any explanation of a dispersal mechanism in the context of the TI-SGTR accident scenario championed by BCOC, the report that is the basis for this concern, *see* Thompson Report, Exh. Schmitz and [Papin], 1999 (Franz Schmitz & Joelle Papin, High burnup effects on fuel behavior under accident conditions: The tests CABRI REP-Na, 270 *Journal of Nuclear Materials* 55 (1999)), is not representative of the circumstances at Shearon Harris. Given BCOC's failure to attempt any dispersion modeling, *see* section II.C.4.a, as contrasted with the Staff's showing regarding its dispersion modeling efforts, *see* Staff Affidavit at 104-06, BCOC's assertions regarding the adequacy of the Staff's dispersion methodology are speculative, at best. Thus, none of these items presents a dispute that warrants further consideration in an evidentiary hearing.

causing a loss of power to the SFP cooling pumps, or unavailability of component cooling water to cool the SFP heat exchangers. *See* Thompson Report at 29, 52. Furthermore, BCOC asserts that these failures are not recoverable, a matter we address more fully regarding event five below.

b. *CP&L Position*

Based on an extensive probabilistic analysis of the loss of fuel pool cooling as a result of the postulated accident, CP&L concluded that the addition of a second (redundant) fuel pool cooling and cleanup system in conjunction with the planned activation of pools C and D actually would reduce the likelihood of a fuel pool cooling failure from what it is for the present pools A and B.

c. *Staff Position*

In analyzing this sequence step, the Staff assessed the probability that the containment failure or containment bypass-related radioactive materials would cause the failure of the component cooling water system, which removes heat from the SFP cooling and cleanup heat exchangers, and failure of the electrical system, thus resulting in a loss of power for SFP cooling and cleanup system pumps. Using information from the IPE for the CDF and applying plant-specific information for internal events, seismic events, and fires, the Staff determined that the overall frequency of events that could lead to an interruption of fuel pool cooling, estimated to be approximately 6.3E-05 per reactor year, is dominated by a loss of offsite power that would affect the operation of the facility's normal and emergency ventilation and exhaust systems. *See* Staff Affidavit at 117-18. The Staff further concluded that the probability of a degraded core accident that leads to an interruption of the SFP cooling function and a containment failure prior to SFP cooling restoration is bounded by 6.3E-06. This determination was based on the Staff's conclusion that the containment failure modes of most concern are the early and late containment failures with a combined probability of 0.1 (10%). *See id.* at 69-71.

d. *Board Analysis*

The Board is seriously troubled by BCOC's claim of certainty — its use of a probability of one — that there will be a loss of SFP cooling as a result of a degraded core accident and containment failure. Putting aside the fact that this claim seemingly ignores the fundamental benefits of engineered safety principles, such as physical separation, redundancy, and diversity in connection with equipment necessary for SFP cooling, the Staff provides persuasive evidence

that the probability of a loss of SPF cooling and makeup is dominated by a loss of offsite power and that there are only limited circumstances after containment failure in which cooling would be lost. Moreover, as is discussed below, the Staff provides a persuasive showing that in many instances credit should be given for the successful recovery of equipment for cooling.

By countering effectively BCOC's argument that the probability of losing SFP cooling is certain (i.e., 1.0) for all accident scenarios, the Staff also counters BCOC's argument that a further evidentiary hearing is warranted relative to this portion of the accident sequence. The Staff's qualitative analysis of the probability of a containment failure or bypass after a degraded core accident is reasonable and supports its conclusion that a containment failure or bypass after a degraded core accident would not have a significant effect in addition to that SFP cooling loss probability that exists for a loss of offsite power.

4. *Event 4 — Extreme Radiation Levels Precluding Personnel Access*

This step in the BCOC postulated sequence assumes that the extreme radiation levels resulting from a reactor containment building breach or bypass precludes access to areas vital to restoring cooling and or makeup water to the fuel pools. *See* Thompson Report at 28-32; CP&L Summary at 57-60; Staff Summary at 31-33; Tr. at 476-79, 484-90, 547-56, 627-30, 637-38, 686, 693, 701-02.

a. BCOC Position

BCOC estimates that as a result of the degraded core and steam generator tube rupture scenarios analyzed, a release of radioactive material through the safety relief valve (SRV) and power operated relief valve (PORV) vent stacks would result in the deposition on the plant site of 5% of the tellurium, 10% of the iodine, and 10% of the cesium radioactive isotopes in the Shearon Harris reactor core within an assumed 200-meter radius centered on the stacks. From this deposition of radioactive material, BCOC calculates dose rates of up to 76 rem per minute outside and up to 110-1100 rem per day during the first release day (300-3000 total for the first 7 days) in the control room and the nearby technical support center (TSC) if there was an offsite power failure that caused an electrical failure to the ventilation systems for these areas after the 4-hour battery backup was exhausted. The control room and TSC are critical areas, according to BCOC, because they are needed for command and communications to coordinate and manage needed activities like maintaining control over the SFP cooling pumps. *See* Thompson Report at 29-32; *id.*, App. E (Radiation Exposure at the Harris Site After an Accident). Given these radiation levels, which would lead to radiation doses to personnel asserted to violate regulatory limits so as to preclude anyone from going into these areas, BCOC also assigns this portion of the sequence a probability of 1.0 (100%). *See id.* at 52.

b. CP&L Position

For the postulated sequence, CP&L calculated radiation levels for areas for which access would be required to assure makeup and cooling to the fuel pools. Using computer modeling of plant thermal hydraulics and the transport of radioactivity, CP&L attempted to determine access, timing, and adverse conditions for critical areas of the plant. These calculations are described in the affidavits of CP&L witnesses Michael J. DeVoe and Benjamin W. Morgan. *See* CP&L Summary, Exhs. 6-7 (Affidavit of Michael J. DeVoe (Nov. 15, 2000); Affidavit of Benjamin W. Morgan (Nov. 15, 2000) [hereinafter Morgan Affidavit]). This information was, in turn, used as the basis for calculating access times based on radiation fields in the following event, for which CP&L provided an overall probability estimate.

c. Staff Position

The Staff performed a detailed qualitative assessment of the impact of radioactive material releases from the postulated sequence on accessibility to critical areas of the reactor auxiliary building (RAB) and the fuel handling building (FHB) needed to assure makeup and cooling water to the pools. The Staff used information on plant layout, expected meteorologic probabilities, and the consequences of the postulated accident scenarios to analyze the radiological and environmental (i.e., steam and heat) conditions at areas of the plant where expected remedial action would be required. This information was drawn from various sections of the Shearon Harris FSAR and Staff reports prepared for this litigation. *See* Staff Affidavit, Exhs. 15 ([CP&L] Response to NRC Staff's First Set of Interrogatories Directed to [CP&L] Regarding Contention EC-6 (Sept. 26, 2000)), 20 (Shearon Harris FSAR, ch. 9), 58 (Shearon Harris FSAR, ch. 12), 63 (Stephen F. LaVie, Staff Analysis of Harris Site Meteorology (Nov. 2000)), 65 (Stephen F. LaVie, Staff Analysis of Radioactivity Release Due to [SFP] Boiling (Nov. 2000)), 72 (Stephen F. LaVie, Staff Analysis of Post-Accident Ground Deposition Dose Rate (Nov. 2000)) [hereinafter Staff Ground Deposition Analysis]. In its detailed review, the Staff considered direct shine from the containment building, direct shine from accident-generated radioactivity in piping systems outside containment, radioactive material in the air of the RAB and the FHB, and radiation from uncovered fuel in the FHB to calculate radiation fields expected to be encountered at various times after the accident and after containment failure or bypass by personnel attempting to restore fuel

pool cooling. The Staff also calculated radiation fields at various FHB access points separated by varying distance and direction from expected accident release points. *See* Staff Affidavit at 98-99. The Staff further considered the historical meteorologic probabilities as to which direction the wind would blow the plume from the release points. The Staff concluded that the FHB access points in relation to expected release points made it unlikely that plume fallout from a breach or bypass would affect all available access points so as to totally preclude access. *See* Staff Ground Deposition Analysis at iii.

d. Board Analysis

BCOC did not perform the detailed calculations of expected radiation fields in various areas of the Shearon Harris plant to which access is needed to restore fuel pool cooling. *See* Tr. at 686. As a consequence, the upshot of its efforts — a simplistic determination that a fixed amount of radioactive material will deposit uniformly in a 200-meter circle centered on the plant SRV and PORV stacks — is unrealistically conservative and lacks a reasonable scientific basis by failing, as it does, to account for building and equipment configuration, historical meteorological data, and accident scenarios. On the other hand, Staff expert Stephen F. LaVie, who has significant experience and training in such calculations, *see* Staff Affidavit at 2; *id.*, Exh. 2 (Resume of Stephen F. LaVie), has provided a credible explanation about the time-dependent, post-accident radiological environment both within and external to the FHB from which access times available to restart fuel pool cooling or makeup can be calculated. Certainly, we find no basis in the information provided by BCOC to convene an evidentiary hearing relative to this segment of the postulated sequence.

5. Event 5 — Inability to Restart Cooling or Makeup Systems Due to Extreme Radiation Doses

This step in BCOC's postulated sequence assumes that CP&L will be unable to recover SFP cooling because the extreme radiation levels from the material escaping from the reactor building precludes plant staff from restoring SFP cooling and makeup water. *See* Thompson Report at 32-38; CP&L Summary at 57-60; Staff Summary at 31-35; Tr. at 490-95, 556-69, 593-94, 630-37, 651-53, 683-84, 694-96, 700-04.

a. BCOC Position

BCOC claims that CP&L cannot use a dose in excess of 5 rem, the maximum permissible occupational dose allowed in 1 year by NRC regulations, 10 C.F.R.

§ 20.1201(a)(1), in planning to recover from an accident. BCOC argues that to use a dose in excess of this value is inappropriate for two reasons. First, according to BCOC, doses in excess of 5 rem can be foreseen and therefore are not covered by the United States Environmental Protection Agency (EPA) protective action guideline (PAG) 2.5 allowing doses of up to 25 rem for life saving and protection of large populations. In addition, BCOC argues that workers will not accept such doses in an emergency. According to BCOC, the radiation field it calculates from the postulated accident exposes personnel in the control room and the TSC to radiation exposures in excess of the 5-rem per year dose limit of section 20.1201(a)(1) and General Design Criterion (GDC) 19, 10 C.F.R. Part 50, App. A, Criterion 19, making the control room uninhabitable for a period in excess of 7 days. This, in turn, would lead to the collapse of the Harris plant command structure and preclude access to areas needed to control SFP cooling. Moreover, BCOC declares this would be exacerbated by the fact that areas outside and inside the RAB would be inaccessible to personnel because of the accident-generated harsh radiation environment and the certainty that electric power likewise would be interrupted for the period that the command structure was inoperative, i.e., in excess of 7 days. Finally, BCOC maintains that all the options required to provide cooling or makeup to the fuel pools require human intervention and such actions would be precluded because of the extreme radiation levels in and around the plant, thus leading to the conclusion that, once again, this portion of the sequence should be assigned a probability of 1.0 (100%). *See* Thompson Report at 32-38, 52; *id.*, App. F (Radiation Exposure: Health Effects and Regulatory Limits).

b. CP&L Position

CP&L expert Benjamin Morgan calculated accessibility to in-plant areas and areas outside the plant buildings using industry-accepted computer codes and NRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." Mr. Morgan indicated the results of these calculations show that various areas of the plant to which access would be necessary after the postulated accident would be reachable to perform activities to provide SFP cooling or makeup. *See* Morgan Affidavit at 4-10; *see also id.*, attachs. B-C (In-Plant Dose Calculation Results; Environmental Dose Calculation Results). Further, CP&L asserts that BCOC misinterprets both NRC regulations and EPA PAG 2.5 relative to worker doses and maintains that the CP&L analysis is consistent with the EPA 25-rem PAG. *See* Tr. at 593-94.

c. Staff Position

Based on the assumption that from the beginning of the accident sequence SFP cooling recirculation is unavailable, the Staff also provided an analysis of the time available for recovery activities before the water in the SFPs boils so as to lower the water level to the top of the fuel storage racks such that makeup is required. *See* Staff Affidavit at 62-69. According to the Staff, 15 days would be available for recovery for pools A and B and 10 days for pools C and D. The Staff then analyzed the alternative methods to provide for pool cooling or makeup. *See id.* at 72-78. Finally, the Staff determined response personnel stay times in the various areas specifically required to recover SFP cooling or makeup functions. These stay times were based on the EPA-sanctioned PAG 2.5 permissible dose of up to 25 rem, which the Staff declares is appropriate under NRC requirements and this EPA guidance, *see* Tr. at 633-37, and the radiation fields determined by the Staff at various locations, as outlined in section II.C.4.c above. *See* Staff Affidavit at 79-111. From this analysis, the Staff determined that there would be options for access to provide makeup or cooling to the pools. *See id.* at 111. Further, the Staff assessed the likelihood of successful operational activities using such access by utilizing a Human Reliability Analysis (HRA) methodology and concluded that, once the makeup method decision was made, the likelihood of success in achieving makeup was high. *See* Staff Affidavit at 111-16. Notwithstanding this conclusion, albeit noting that no HRA methodology has been constructed to provide human error probabilities for such recovery situations, the Staff nonetheless assigned what is described as a conservative probability of 0.1 (10%) that the SFP cooling restoration or makeup would not be successful. *See* Staff Affidavit at 116-17. Finally, the Staff agrees with CP&L that BCOC misinterprets both NRC regulations, including the agency's emergency planning response requirements, *see* 10 C.F.R. § 50.47(b)(11), and EPA PAG 2.5 relative to worker doses and asserts that the Staff's analysis is consistent with the EPA 25-rem PAG. *See* Tr. at 630-37.

d. Board Analysis

Considering first the question of the maximum allowable dose to be used in calculating whether access can be effected in an emergency situation, it is clear to us from a review of the applicable regulatory provisions — 10 C.F.R. §§ 20.1001(b), 20.1201(a)(1), 50.47(b)(11) — that there is no regulatory bar that prohibits CP&L from using a 25-rem dose limit in an actual emergency or in planning a response to such an emergency to assure SFP cooling after an accident.[6] Likewise, EPA PAG 2.5 clearly allows a dose of up to 25 rem for life saving and protection of large populations. *See* Staff Affidavit, Exh. 55, at 2-9 to -11 (EPA, Manual for Protective Action Guides and Protective Actions for Nuclear Incidents (May 1992)).[7] Moreover, because this dose is within regulatory standards, the Board will not engage in the unsupported surmise, as BCOC would have us do, that knowledgeable plant personnel would be wholly unwilling to accept such doses in an emergency such as the postulated accident sequence. The Board thus concludes that it is appropriate to use a permissible dose of 25 rem for purposes of calculating stay times and probabilities that personnel at the plant would be able to perform the necessary activities required to restore SFP cooling and makeup.

As noted above, using the calculated radiation fields and the 25-rem person dose, the Staff calculated times available to perform SFP cooling and makeup restoration activities for the various alternative methods of providing makeup or cooling to the SFPs. *See* Staff Affidavit at 109 (Table 2, Makeup Alternatives). The Staff's analysis in support of its probability estimate, which is supported by CP&L's detailed evaluation, appears reasonably thorough and credible based on existing regulations and guidance for exposure to emergency workers, as well as on the expected radiation fields in locations at which SFP cooling recovery actions must take place and the availability of various alternative sources of cooling water. In contrast, BCOC provides us with no credible analysis, other than its unsupported assertion about uniform radioactive materials disposition and its mistaken interpretation of NRC requirements and EPA's PAG 2.5, to support its conclusion that any access to areas of the plant needed for SFP recovery and makeup would be precluded by high radiation fields.[8] Once again, we find nothing relative to this sequence event that establishes the need for an evidentiary hearing.

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[6] In this regard, unlike BCOC, *see* Thompson Report at 33 & n.64, in the context of reviewing what clearly are low probability accident scenarios, we do not equate consideration of radiation exposure in the course of doing a probability analysis with "forseeability" relative to the EPA PAG so as to mandate application of a 5-rem exposure limit.

[7] In pertinent part, this EPA PAG provides:

> Doses to all workers during emergencies should, to the extent practicable, be limited to 5 rem. There are some emergency situations, however, for which higher exposure limits may be justified. Justification of any such exposure must include the presence of conditions that prevent the rotation of workers or other commonly-used dose reduction methods. Except as noted below, the dose resulting from such emergency exposure should be limited to 10 rem for protecting valuable property, and to 25 rem for life saving activities and the protection of large populations. In the context of this guidance, exposure of workers that is incurred for the protection of large populations may be considered justified for situations in which the collective dose avoided by the emergency operation is significantly larger than that incurred by the workers involved.

Staff Affidavit, Exh. 55, at 2-11.

[8] Relative to this event, BCOC also makes the assertion that a purported Staff failure to make any assessment of the probability of restoring cooling provides a litigable dispute, *see* Tr. at 483, a claim that we find wholly without merit or worthy of further consideration in an evidentiary hearing given the discussion above regarding the Staff's analysis. The same is true of BCOC's claim of a factual dispute regarding firefighter access to a 195 degree Fahrenheit (°F) steam environment in the FHB, *see* Tr. at 494-95, which does not account for existing firefighter training, *see* CP&L Summary, Exh. 5, at 10 (temperatures in range of 300°F not unusual during fire brigade training sessions) (Affidavit of Eric A. McCartney).

6. *Event 6 — Loss of Most or All Pool Water Through Evaporation*

At this step of the postulated accident sequence, all of the water covering the spent fuel is assumed lost by evaporation because cooling or makeup water could not be restored. *See* Thompson Report at 39-40; CP&L Summary at 60-62; Staff Summary at 33-34; Tr. at 495-97, 560-64, 638.

a. BCOC Position

BCOC asserts that with the loss of SFP cooling capability after the postulated accident, boiling would occur in the pools to such an extent that the water level would reach the top of the fuel in pool A in a period of 4.7 days and pools C and D in a period of 10.2 to 116 days, depending on the heat load in the pool. BCOC also contends that this would happen with certainty — a probability of 1.0 or 100% — because the high radiation fields described in section II.C.4.a above would preclude any recovery of cooling or makeup systems to the pools. And this loss of water in pool A, BCOC declares, would result in an exothermic oxidation reaction that would release radioactive material in and around the FHB. *See* Thompson Report at 39-40.

b. CP&L Position

CP&L calculates it will take more than 8 days to uncover fuel in pools A and B and almost 100 days to uncover fuel in pools C and D. *See* Burns Affidavit at 11-12. It is unlikely this would ever happen, according to CP&L, because there are many ways to establish makeup and cooling to the SFPs, possibilities that will be enhanced by the redundant SFP cooling and cleanup system for pools C and D that provide additional pathways for makeup water injection. *See* ERIN Report at A-28 to -30 (Table A-1, [SFP] Makeup). CP&L concluded that at least one makeup water lineup was possible within 4 days for all the accident-initiating sequences of the postulated core damage accident. *See* Burns Affidavit at 12.

c. Staff Position

The Staff likewise analyzed the probability of success in restoring cooling and makeup water to the SFPs after the postulated accident and containment failure or bypass. For a late containment failure scenario — i.e., with failure at 90 hours — the Staff concluded there was a high probability of success in restoring cooling. The control room would be habitable for most of the period and alarms would indicate pool cooling failure and level reductions such that the plant staff could respond in a timely manner. *See* Staff Affidavit at 111-14. For an early containment failure scenario, the Staff assumed that although the

control room would not be habitable, command and control would be available in the TSC and/or the NRC incident response center. Moreover, citing NRC emergency operations center guidance regarding post-accident SFP cooling, the Staff asserted it would be unreasonable to assume there was any likelihood after the postulated accident that SFP cooling would be forgotten or ignored. *See id.* at 114. Additionally, the Staff reviewed the methods required by CP&L plant staff to restore cooling or initiate makeup and determined that there is a high likelihood of success in obtaining access and performing the necessary functions to restore cooling or makeup. As was noted previously in section II.C.5.c, the Staff assigned a probability of no greater than 0.1 (10%) that such actions would be unsuccessful. *See* Staff Affidavit at 116-17.

d. Board Analysis

As we have already noted, the Board adopts the Staff's analysis regarding CP&L's ability to provide SFP makeup and cooling. As we discussed in section II.C.5.d above, the Staff calculated reasonable stay times for the many SFP cooling and makeup methods. Even if CP&L loses the ability to run the plant from the control room, there are procedures in place for both CP&L and the NRC to exercise command and control to make decisions about safeguarding SFP cooling integrity. Putting aside the relatively low makeup water flow rates that likely would be needed, there are myriad ways to get the recovery makeup water into the fuel pools, which are not adequately accounted for in BCOC's assignment of a certainty to this step of the sequence. Ultimately, nothing presented by BCOC establishes the need to proceed to an evidentiary hearing on this aspect of the postulated scenario.

7. *Event 7 — Initiation of an Exothermic Oxidation Reaction in Pools C and D*

At this final step of BCOC's postulated accident sequence, the spent fuel cladding spontaneously ignites after the cooling water is lost by evaporation as a result of steps one through six above. Such a reaction essentially means that the fuel rapidly oxidizes (i.e., burns) and releases high levels of radioactive material into the environment around the Shearon Harris plant site. *See* Thompson Report at 40-42; CP&L Summary at 65-68; Staff Summary at 6; Tr. at 497-99, 564-67, 596-97, 639-41, 694.

a. Parties' Position

This last step of BCOC's postulated sequence looks to the probability that an exothermic oxidation reaction would occur in the pools after the fuel pool

cooling water evaporates and the fuel is uncovered. All parties agreed for the purpose of this analysis to assume that an exothermic reaction would take place. Although CP&L and the Staff are skeptical that such a reaction would take place with certainty, particularly if evaporation of the fuel pooling water occurred in a pool containing only aged spent fuel, *see* CP&L Summary at 67; Staff Affidavit at 124, they both accept for purposes of the analysis that an exothermic oxidation reaction would occur in pools C and D with certainty, i.e., with a probability of 1.0 (100%).

b. Board Analysis

The Board accepts that there is no controversy among the parties associated with this event in BCOC's postulated accident sequence. As such it does not provide a basis for further evidentiary hearings.

8. Cumulative Scenario Probability

As a result of its analysis of the contention EC-6 accident sequence, BCOC provides a probability of 1.6E-05 per reactor year as its best estimate of the overall probability of an oxidation reaction in pools C and D. *See* BCOC Summary at 40. CP&L's best estimate is 2.7E-08. *See* CP&L Summary at 51. The Staff provides a best estimate of the overall probability of the postulated accident scenario as 2.0E-07 per reactor year. *See* Staff Summary at 44. The Board's summary of the overall cumulative probabilities (per reactor year) determined by each of the parties for BCOC's postulated accident sequence is presented in the table below. The cumulative probability at step N (S_N) is defined as the product of the probability of all the preceding steps up to and including step N, namely ($S_N = P_1 P_2 P_3 \ldots P_N$, where P_N is the individual probability for step N.[9]

[9] In this context, S_1 represents the probability of occurrence of step one of the postulated accident sequence. S_2 represents the probability of the occurrence of step one and step two of the scenario. Finally, S_7 represents the probability of occurrence of the entire seven-event accident sequence.

BCOC Contention EC-6 Accident Scenario Cumulative Probability (S_N)				
Sequence Event (N)		**BCOC S_N**	**CP&L S_N**	**Staff S_N**
1	Degraded core accident	3.1E-05		1.2E-04
2	Containment failure or bypass	1.6E-05	7.7E-06[a]	
3	Loss of SFP Cooling and/or Makeup Loss	1.6E-05		6.3E-06[c]
4	Radiation Dose Precludes Access	1.6E-05		
5	Inability to restart SFP cooling	1.6E-05		2.0E-07[d]
6	Loss of part or all of SFP water by evaporation	1.6E-05	2.7E-08[b]	2.0E-07
7	Initiation of an exothermic oxidation reaction in Pools C and D	1.6E-05	2.7E-08	2.0E-07
Overall Sequence Probability (per reactor year)		**1.6E-05**	**2.7E-08**	**2.0E-07**

[a] CP&L combined its analysis of the first two steps.
[b] CP&L combined its analysis of steps three through six.
[c] Staff combined its analysis of steps two and three.
[d] Staff combined its analysis of steps four and five.

Relative to these estimates, for the reasons set forth in sections II.C.1 through II.C.7 above, the Board concludes that the overall probability of the BCOC postulated accident sequence resulting in an exothermic oxidation reaction in the Harris plant SFPs is conservatively in the range described by the Staff: 2.0E-07 per reactor year (two occurrences in 10 million reactor years) or less.

D. Cumulative Scenario Probability as Remote and Speculative

With this probability figure before us, we must next consider whether it appropriately can be characterized as "remote and speculative" within the meaning of NEPA, so as to provide a substantive basis for the BCOC challenge to the Staff's EA determination. Citing agency consideration of severe accident probability estimates for reactor-related internal and/or external events in various NEPA or Atomic Energy Act contexts, both CP&L and the Staff assert that probabilities on the order of at least 1E-06 (one in one million) should be considered remote and speculative for NEPA purposes. *See* CP&L Summary at 46-50 (citing, e.g., SECY-98-231, Authorization of the Trojan Reactor Vessel Package for One-time Shipment for Disposal (Oct. 2, 1998) (1E-06); NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants § 5.2.3.1 (Supp. 2 1999) (8.9E-05)); Staff Summary at 36-43 (citing, e.g., *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-877, 26 NRC 287, 293 (1987) (3E-05 to 1E-10); *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 2), ALAB-603, 12 NRC 30, 45 (1980) (1E-06 to 1E-07); *Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), LBP-78-15, 7 NRC 642, 699 (1978) (1E-06));

see also Tr. at 605-12, 659-61. BCOC, on the other hand, while suggesting that its probability estimate of 1.6E-05 is sufficient to establish that the contention EC-6 accident sequence is not remote and speculative, also declares that several factors should counsel serious Board concern about whether, in this proceeding, lower probability estimates should be considered as not falling within the category of remote and speculative. Among these are the need to take a "hard look" at potential environmental consequences in an EA; the level of uncertainty that is involved in the probability analyses used to support the EA determination; and particular uncertainty factors such as the use of unverifiable judgments rather than calculations to account for unknown aspects of plant behavior, the degree to which acts of malice, gross design errors, unforseen accident sequences or phenomena, or degraded operation standards could influence those probability analyses, and dependance on new and untested applications of PRA techniques. *See* BCOC Summary at 23-30; *see also* Tr. at 499-508.

Notwithstanding the suggestion that we draw a "line in the sand" by declaring "remote and speculative" those matters whose probabilities fall into the range of 1E-06 or lower, in the context of this proceeding we need do no more than determine whether the Staff's 2.0E-07 per reactor year probability analysis estimate that we find compelling falls beyond that line. The various agency determinations cited by CP&L and the Staff indicate that this estimate falls within the category of remote and speculative matters, assuming we do not consider the BCOC concerns described above sufficient to remove this estimate from that category.

In this regard, we note that whatever may have been the case previously, the information submitted by the Staff in its section 2.1113 written presentation regarding contention EC-6 makes it readily apparent that, relative to its EA determination, any requisite "hard look" has been taken at this point. *See* 10 C.F.R. § 51.34(b). And concerning the matter of probability analysis uncertainty, BCOC has not presented any specific information other than its expressed concerns about the reliability of the probability analysis process used in addressing its contention EC-6, particularly the purported lack of "peer review." *See* BCOC Summary at 28, 29; Tr. at 499-501, 507-08, 514, 686-89.

Dr. Thompson apparently was the sole contributor to BCOC's position. *See* Thompson Report at 2. No peer review of Dr. Thompson's work was performed. *See* Tr. at 524. In contrast, CP&L and the Staff both attest to a peer review-type process in connection with their analyses.[10] In connection with

[10] According to CP&L, to answer the "best estimate" question posed by the Board in LBP-00-19, with considerable assistance from outside contractor ERIN Engineering and Research, Inc., CP&L sought to obtain a probability analysis of BCOC's postulated accident sequence. In doing so, ERIN reviewed and utilized existing plant-specific information, including the Shearon Harris PSA and IPEEE, which were not prepared by ERIN and are in accord with NRC Generic Letter 88-20, to provide an updated Shearon Harris PSA. This work, in turn, was reviewed by CP&L personnel and ERIN personnel who were not members of the immediate team performing the analysis. Moreover, CP&L declares that its contractor was hired to answer the Board's questions, not to satisfy its client CP&L. *See* CP&L Summary at 52-53; Tr. at 540-41, 569-71, 585-88, 595-96, 690-92.

the Staff's submission, which we have explained in section II.C, we consider an appropriate probability analysis tool in this instance,[11] the Staff confirms that the key documents it used — the CP&L 1993 IPE, 1995 IPEEE, and 1995 PSA — were subject to peer review when created. In addition, the IPE and IPEEE were reviewed by the Staff independent of this proceeding. Moreover, the Staff's analysis of the key elements of the contention EC-6 scenario had internal peer or supervisory review: the Staff fielded a panel of risk analysis practitioners from various disciplines to prepare its position, which was then subjected to a peer review by employees from the agency's Office of Nuclear Regulatory Research. *See* Staff Summary at 25-26, 34-35; Staff Affidavit at 9, 15, 16-17, 122; Tr. at 644-45.

The Board recognizes that, consistent with the Commission's guidance in its *Vermont Yankee* opinion "that future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual probabilities and accident scenarios being analyzed," CLI-90-7, 32 NRC at 132, we must have a significant degree of confidence in the reliability of the analyses we receive from the parties. At the same time, we do not think necessary, and did not request that the parties provide, a new, detailed PRA analysis relative to the contention EC-6 accident scenario. As was noted above, all of the parties began their evaluations of the postulated sequence with the CP&L PSA and/or IPE or IPEEE that have undergone peer review. Further, in the absence of any specific evidence of bias or mistake, the subsequent internal review of the components of its contention EC-6 probability analysis by Staff senior technical or supervisory personnel who were not involved in preparing the Staff's analysis is adequate in this context to provide the Board with confidence in the reliability of the Staff analysis regarding all of the important issues associated with each step of the postulated sequence. *Cf. United States v. Chemical Foundation, Inc.*, 272 U.S. 1, 14-15 (1926) (presumed that government official can be expected faithfully to execute his or her official duties). Thus, the fact that the peer review process for the Staff's contention EC-6 probability analysis may not be fully in accord with BCOC's criteria of complete independence is not a disqualifying factor, or one that mandates further evidentiary proceedings.

E. Additional Board Questions

As was noted in section II.B above, the Board also asked for party responses to two additional questions regarding (1) the relevance of a 2E-06 per reactor year estimate in NUREG-1353, a 1989 Staff generic study of SFP design basis accidents, and concerns about exothermic reactions expressed in an April 13,

[11] In large part, BCOC's uncertainty concerns relate to the CP&L PRA analysis in the ERIN Report rather than the Staff's analysis. *See* Tr. at 678-83.

2000 ACRS letter; and (2) the required scope of any EIS, if one is found to be necessary.

Regarding the first of these two Board inquiries, BCOC questions the relevance of NUREG-1353 because that report did not consider the seven-step sequence being examined under contention EC-6 and because the SFP conditions assumed in NUREG-1353 are not representative of the Shearon Harris rack configuration and fuel loading characteristics. With regard to the April 2000 ACRS letter, BCOC notes that although the February 2000 Staff draft technical study on SFP accident risk at decommissioning plants that is the subject of the ACRS letter did not address partial drainage or fuel/rack relocation heat transfer implications, it did acknowledge the limitations of previous analyses relative to exothermic reactions. *See* BCOC Summary at 40; Thompson Report at 44-46. Both CP&L and the Staff likewise declare NUREG-1353 has no direct relevance to the individual events in the scenario since that report uses a high ground acceleration earthquake rather than severe core damage accidents as an initiating event. And with regard to the April 2000 ACRS letter, CP&L and the Staff assert that the exothermic reaction concern that is the focus of that letter is irrelevant because it has been assigned a probability of 1.0 (100%) in scenario event seven. *See* CP&L Summary at 73-77; Staff Summary at 44-46; Tr. at 612-13.

After reviewing the arguments of the parties regarding this question, the Board agrees that NUREG-1353 has no direct relevance to our resolution of BCOC contention EC-6. The assignment of a probability of 1.0 to scenario step seven has incorporated the concerns raised in connection with the April 2000 ACRS letter as well.[12] And neither, of course, provides cause for further evidentiary proceedings.

Finally, given our disposition of this proceeding, the EIS-scope matter posed in the final question does not provide grounds for an evidentiary hearing or, indeed, warrant further consideration in this proceeding.

III. CONCLUSION

Based on the record before us, pursuant to 10 C.F.R. § 2.1115, we conclude Intervenor BCOC has failed to demonstrate relative to its contention EC-6 challenge to CP&L's December 1998 Harris facility SFP expansion amendment request, that there is any genuine and substantial dispute of fact or law that only can be resolved with sufficient accuracy in an evidentiary hearing. At the same time, we find the Staff has demonstrated the sufficiency of its analysis, which places the overall probability that the accident sequence postulated under BCOC contention EC-6 will result in an exothermic oxidation reaction in the Harris facility SFPs conservatively in the range of 2.0E-07 per reactor year or less. As a result, the Staff has met its burden to establish that such a scenario can properly be characterized as "remote and speculative" so as not to warrant preparation of an EIS regarding CP&L's amendment application.

We thus dispose of this contention by affirming the Staff's December 1999 EA FONSI determination, as supplemented by this decision and the accompanying record and, having resolved the only outstanding matter at issue in this cause, terminate this proceeding.

For the foregoing reasons, it is, this first day of March 2001, ORDERED that:

1. With respect to BCOC contention EC-6, Environmental Impact Statement Required, in accordance with 10 C.F.R. § 2.1115(b), because (a) there is no genuine and substantial dispute of fact or law that only can be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing; and (b) the NRC Staff has established that the accident scenario that is the basis for that issue statement is remote and speculative so as not to warrant the preparation of an EIS, the December 1999 Staff EA FONSI determination relative to the December 1998 CP&L SFP expansion license amendment application is *affirmed*, as supplemented by this decision and the record accompanying it; and,

2. Because there are no remaining disputed issues of fact or law requiring resolution in an adjudicatory hearing and all issues in this proceeding have been resolved in favor of granting the December 1998 license amendment application, the Staff is *authorized* to issue the license amendment requested by CP&L and, pursuant to section 2.1115(a)(2), this proceeding is *dismissed*.

In accordance with 10 C.F.R. §§ 2.760, 2.764, and the Commission's decision in CLI-01-7, 53 NRC 113, 119 (2001), this decision shall become effective immediately. It will constitute the final decision of the Commission forty (40) days from the date of issuance, or on *Tuesday, April 10, 2001*, unless a petition for review is filed in accordance with 10 C.F.R. § 2.786, or the Commission directs otherwise. Within fifteen (15) days after service of this decision, any party may file a petition for review with the Commission on the grounds specified in 10 C.F.R. § 2.786(b)(4). The filing of a petition for review is mandatory for a party to have exhausted its administrative remedies before seeking judicial review. Within ten (10) days after service of a petition for review, any party to the proceeding may file an answer supporting or opposing Commission review.

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[12] Following the Board's December 2000 section 2.1113 oral argument, the agency released the October 2000 final version of the Staff study on SFP accident risks at decommissioning plants in which the Staff concluded that although the risk of an exothermic reaction in the form of a zirconium fire was very low, the radiological effects of such a fire would be serious. *See* Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Oct. 2000), at viii (available at www.nrc.gov/NRC/REACTOR/DECOMMISSION/SF/index.html). Because a probability of 1.0 already has been assigned to the step in the contention EC-6 scenario that postulates an exothermic reaction, this report is not relevant to the matters at issue in this proceeding.

The petition for review and any answers shall conform to the requirements of 10 C.F.R. § 2.786(b)(2)-(3).

THE ATOMIC SAFETY AND
LICENSING BOARD[13]

G. Paul Bollwerk, III, Chairman
ADMINISTRATIVE JUDGE

Dr. Peter S. Lam
ADMINISTRATIVE JUDGE

Thomas D. Murphy
ADMINISTRATIVE JUDGE

Rockville, Maryland
March 1, 2001

[13] Copies of this Memorandum and Order were sent this date by Internet e-mail transmission to counsel for (1) Applicant CP&L, (2) Intervenor BCOC, and (3) the Staff.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD PANEL

Before Administrative Judges:

Ann Marshall Young, Chair
Dr. Charles N. Kelber
Thomas S. Moore

In the Matter of

Docket Nos. 50-336-LA
50-423-LA
(ASLBP No. 00-783-09-LA)

NORTHEAST NUCLEAR ENERGY COMPANY
(Millstone Nuclear Power Station, Units 2 and 3)

March 29, 2001

In this operating license amendment proceeding, the Licensing Board majority finds that the Petitioners have not proffered an admissible contention so the intervention petition must be denied.

RULES OF PRACTICE: CONTENTIONS

The contention pleading criteria set forth in 10 C.F.R. § 2.714(b)(2) are mandatory and must be scrupulously followed. As the Commission has stated with respect to these regulatory provisions, "[i]f any one of these requirements is not met, a contention must be rejected." *Arizona Public Service Co.* (Palo Verde Nuclear Generating Station, Units 1, 2, and 3), CLI-91-12, 34 NRC 149, 155 (1991).

RULES OF PRACTICE: CONTENTIONS

The provisions of 10 C.F.R. § 2.714(b)(2)(i), (ii), and (iii) were specifically added by the Commission "to raise the threshold bar for an admissible

9. *Date:* January 12, 2001.
Time: 9:00 a.m. to 5:00 p.m.
Room: 315.
Program: This meeting will review applications for Fellowship Programs at Independent Research Institutions in Collaborative Research, submitted to the Division of Research Programs at the September 1, 2000 deadline.

Laura S. Nelson,
Advisory Committee Management Officer.
[FR Doc. 00–33059 Filed 12–27–00 8:45 am]
BILLING CODE 7536–01–M

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–400]

Carolina Power & Light Company; Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 103 to Facility Operating License No. NPF–63 issued to Carolina Power & Light Company (CP&L, the licensee), which revised the Technical Specifications (TS) for operation of the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), located in Wake and Chatham Counties, North Carolina. The amendment is effective as of the date of issuance.

The amendment modified the TS to support a modification to HNP to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) C and D and placing the pools in service. Specifically, the amendment consists of: (1) A revision to TS 5.6 to identify pressurized water reactor fuel burnup restrictions, boiling water reactor fuel 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; (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; and (3) an unreviewed safety question for additional heat load on the CCW system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the **Federal Register** on January 13, 1999 (64 FR 2237). A request for a hearing was filed on February 12, 1999, by the Board of Commissioners of Orange County, North Carolina (BCOC).

On July 12, 1999, the Atomic Safety and Licensing Board (ASLB) ruled that BCOC had standing and had submitted two admissible contentions. The two contentions related to (1) whether General Design Criterion 62 allows the use of administrative controls to prevent criticality (TC–2); and (2) the adequacy of the licensee's proposed alternative plan for the cooling system piping (TC–3). On July 29, 1999, the ASLB granted CP&L's request to hold the hearing in accordance with the hybrid hearing procedures of 10 CFR Part 2, Subpart K. On January 4, 2000, all parties filed written summaries and on January 21, 2000, the ASLB heard oral arguments related to the two admitted contentions. On May 5, 2000, the ASLB issued a decision in favor of CP&L, stating that "(1) there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing; and (2) contentions TC–2 and TC–3 are disposed of as being resolved in favor of CP&L."

On January 31, 2000, BCOC filed four late-filed environmental contentions that challenged the adequacy of the staff's December 21, 1999, environmental assessment related to CP&L's amendment request. On March 3, 2000, the NRC and CP&L responded to the late-filed contentions, and on March 13, 2000, BCOC submitted its reply to the responses. On August 7, 2000, the ASLB issued its Ruling on Late-filed Environmental Contentions. In its ruling, the ASLB admitted one environmental contention (EC–6) regarding the probability of occurrence of BCOC's postulated accident scenario. On November 20, 2000, all parties filed written summaries and on December 7, 2000, the ASLB heard oral arguments related to EC–6.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding or completion of any required hearing, where it has determined that no significant hazards considerations are involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards considerations. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (64 FR 71514).

For further details with respect to the action see (1) the application for amendment dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, (2) Amendment No. 103 to License No. NPF–63, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 21st day of December 2000.

For the Nuclear Regulatory Commission.

Richard P. Correia,
Chief, Section 2, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.
[FR Doc. 00–33152 Filed 12–27–00; 8:45 am]
BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–305]

Nuclear Management Company, LLC; Notice of Withdrawal of Application for Amendment to Facility Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has granted the request of Nuclear Management Company, LLC (the licensee) to withdraw the June 7, 1999, as supplemented February 4, and

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 21, 2000

Mr. James Scarola, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT RE: EXPANSION OF SPENT FUEL STORAGE CAPACITY (TAC NO. MA4432)

Dear Mr. Scarola:

The Nuclear Regulatory Commission has issued Amendment No. 103 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit No. 1 (HNP), in response to your request dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000. This amendment supports a modification to HNP to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) C and D and placing the pools in service. Specifically, this amendment: 1) revises Technical Specification 5.6 to identify pressurized water reactor fuel burnup restrictions, boiling water reactor fuel 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; and 2) resolves an unreviewed safety question for additional heat load on the component cooling water (CCW) system.

In addition, your December 23, 1998, submittal included 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 CCW and SFPs C and D cooling and cleanup system piping. The staff has completed its review of your alternative plan, as documented in sections 3.7 and 3.8 of the enclosed Safety Evaluation. The staff has determined that the proposed alternative for the missing weld documentation for SFPs C and D piping would provide an acceptable level of quality and safety. Accordingly, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i).

A copy of the Notice of Issuance and Final Determination of No Significant Hazards Consideration, which is being sent to the Office of the Federal Register for publication, is also enclosed.

Sincerely,

Richard J. Laufer

Richard J. Laufer, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 103 to NPF-63
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Carolina Power & Light Company

Shearon Harris Nuclear Power Plant
Unit 1

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. NPF-63

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 103 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Additionally, the license is amended to authorize revision of the Final Safety Analysis Report (FSAR) to reflect the change in the minimum specified component cooling water system flow to the residual heat removal system heat exchanger from 5600 gpm to 5200 gpm. The licensee shall make this update to the FSAR, as authorized by this amendment, in accordance with 10 CFR 50.71(e).

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 21, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
xvii	xvii
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5-7	5-7
----	5-7a
----	5-7b

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DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 except that limited substitution of fuel rods by filler rods consisting of Zircaloy-4, stainless steel, or by vacancies may be made in fuel assemblies if justified by a cycle-specific evaluation. Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235. Fuel with enrichments greater than 4.20 weight percent U-235 shall contain sufficient integral burnable absorbers such that the requirement of Specification 5.6.1.1.b is met.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9410 ± 100 cubic feet at a nominal T_{avg} of 580.8°F.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with a k_{eff} less than or equal to 0.95 when flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3.2.6 of the FSAR.

1. The reactivity margin is assured for pools "A" and "B" by maintaining:

   a. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the flux trap style PWR storage racks and 6.25 inch center-to-center distance in the BWR storage racks.

   b. The maximum core geometry K_∞ for PWR fuel assemblies less than or equal to 1.470 at 68°F.

2. The reactivity margin is assured for pools "C" and "D" by maintaining a nominal 9.017 inch center-to-center distance between fuel assemblies placed in the non-flux trap style PWR storage racks and 6.25 inch center to center distance in the BWR storage racks. The following restrictions are also imposed through administrative controls:

   a. PWR assemblies must be within the "acceptable range" of the burnup restrictions shown in Figure 5.6-1 prior to storage in Pools "C" or "D".

   b. BWR assemblies are acceptable for storage in Pool "C" provided the maximum planar average enrichments are less than 4.6 wt% U235 and K_{inf} is less than or equal to 1.32 for the standard cold core geometry (SCCG).

DRAINAGE

5.6.2 The pools "A", "B", "C" and "D" are designed and shall be maintained to prevent inadvertent draining of the pools below elevation 277.

CAPACITY

5.6.3.a Pool "A" contains six (6 x 10 cell) flux trap type PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool "B" contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) flux trap style PWR racks and seventeen (11 x 11 cell) BWR racks and is licensed for one additional (11 x 11 cell) BWR rack that will be installed as needed. The combined pool "A" and "B" licensed storage capacity is 3669 assemblies.

5.6.3.b Pool "C" is designed to contain a combination of PWR and BWR assemblies. Pool "C" can contain two (11 x 9 cell) and nine (9 x 9 cell) PWR racks for storage of 927 PWR assemblies. Pool "C" can contain two (8 x 13 cell), two (8 x 11 cell), six (13 x 11 cell), and nine (13 x 13 cell) BWR racks for storage of 2763 BWR assemblies. The (9 x 9 cell) PWR racks and the (13 x 13 cell) BWR racks are dimensioned to allow interchangeability between PWR or BWR storage rack styles as required. The racks in pool "C" will be installed as needed.

5.6.3.c Pool "D" contains a variable number of PWR storage spaces. These racks will be installed as needed. Pool "D" is designed for a maximum storage capacity of 1025 PWR assemblies.

5.6.3.d The heat load from fuel stored in Pools "C" and "D" shall not exceed 1.0 MBtu/hr.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

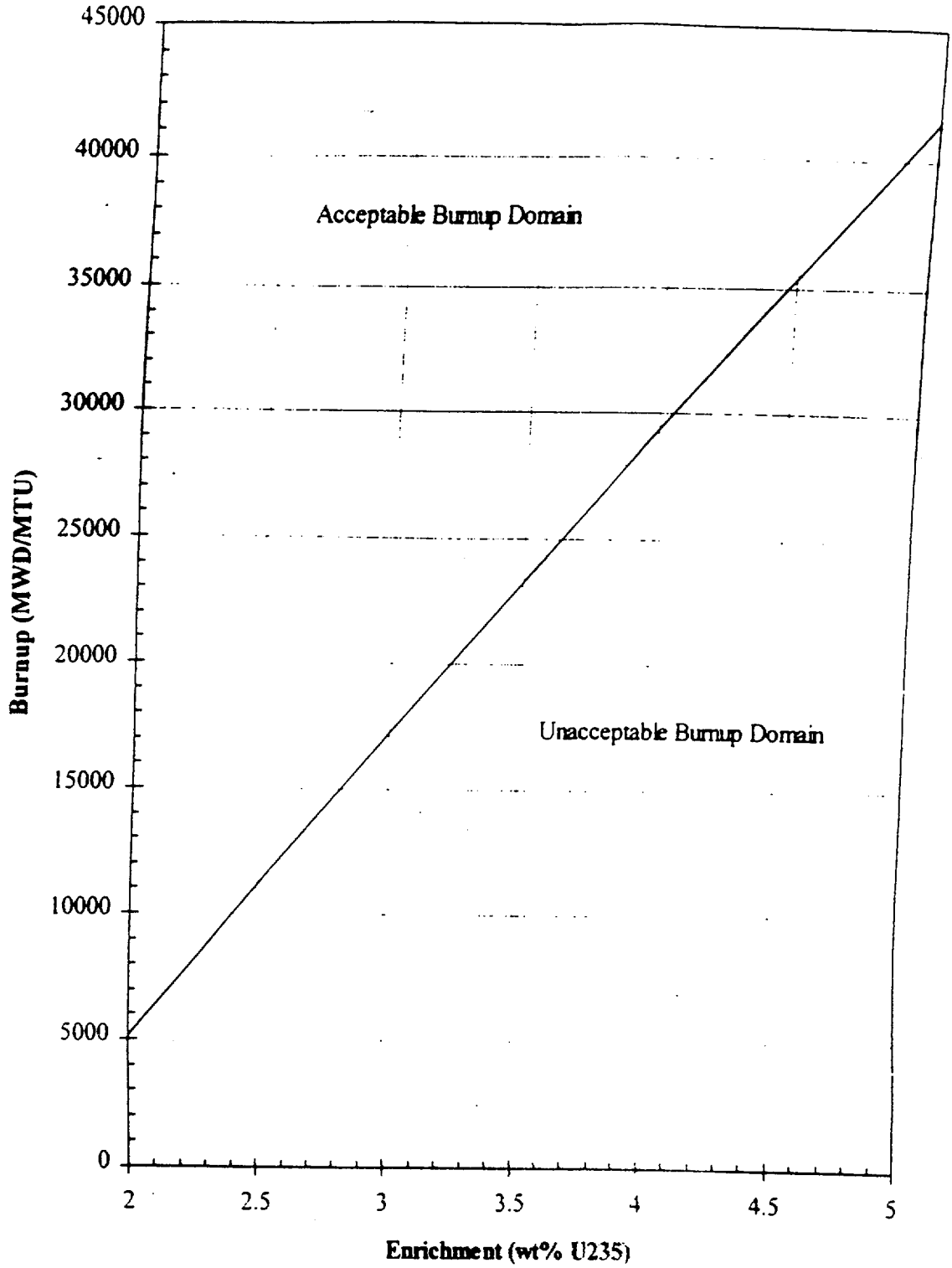

FIGURE 5.6-1
BURNUP VERSUS ENRICHMENT FOR PWR FUEL

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

1.1 License Amendment Request

By letter dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000 (references 1 - 11), Carolina Power and Light Company (CP&L, the licensee), requested a change to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed amendment would support a modification to HNP to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) C and D and placing the pools in service. Specifically, the proposed action consists of: 1) a revision to TS 5.6 to identify pressurized water reactor (PWR) fuel burnup restrictions, boiling water reactor (BWR) fuel 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; 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; and 3) an unreviewed safety question (USQ) for additional heat load on the CCW system.

The supplemental submittals dated March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, provided clarifying information that did not change the scope of the original *Federal Register* notice published on January 13, 1999 (64 FR 2237).

1.2 Atomic Safety and Licensing Board Hearing

In response to the January 13, 1999, *Federal Register* Notice, on February 12, 1999, the Board of Commissioners of Orange County North Carolina (BCOC) filed a Request for a Hearing and Petition to Intervene in the license amendment proceeding (ref. 23). Subsequently, on April 5, 1999, BCOC filed eight contentions (three technical and five environmental) (ref. 24). The NRC staff and the licensee provided responses to BCOC's contentions on May 5, 1999 (ref. 25, 26). The Atomic Safety and Licensing Board (ASLB) held a pre-hearing conference in Chapel Hill, North Carolina on May 13, 1999 (ref. 27).

On July 12, 1999, the ASLB issued its Ruling on Standing and Contentions (ref. 28). In its ruling, the ASLB stated that BCOC had standing and had submitted two admissible contentions. The two contentions related to (1) whether General Design Criterion (GDC) 62 allows the use of administrative controls to prevent criticality (TC-2); and (2) the adequacy of the licensee's proposed alternative plan for the cooling system piping (TC-3). On July 29, 1999, the ASLB granted CP&L's request to hold the hearing in accordance with the hybrid hearing procedures of 10 CFR Part 2, Subpart K (ref. 29).

On January 4, 2000, all parties filed written summaries (ref. 30, 31, 32) and on January 21, 2000, the ASLB heard oral arguments related to the two admitted contentions (ref. 33). On May 5, 2000, the ASLB issued a decision in favor of CP&L (ref. 34). In its decision, the ASLB concluded that "(1) there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing; and (2) contentions TC-2 and TC-3 are disposed of as being resolved in favor of CP&L."

On January 31, 2000, BCOC filed four late-filed environmental contentions (ref. 35), which challenged the adequacy of the staff's December 21, 1999, environmental assessment (ref. 36) related to CP&L's amendment request. On March 3, 2000, the NRC and CP&L responded to the late-filed contentions (ref. 37, 38), and on March 13, 2000, BCOC submitted its reply to the responses (ref. 39). On August 7, 2000, the ASLB issued its Ruling on Late-filed Environmental Contentions (ref. 44). In its ruling, the ASLB admitted one environmental contention (EC-6) regarding the probability of occurrence of BCOC's postulated accident scenario. On November 20, 2000, all parties filed written summaries (ref. 45, 46, 47) and on December 7, 2000, the ASLB heard oral arguments related to EC-6.

On May 22, 2000, BCOC filed a petition (ref. 40) for the Commission to review the ASLB's May 5, 2000, decision (ref. 34) in the Harris Nuclear Plant spent fuel storage case. BCOC stated in its petition that "The Commission should take review of clearly erroneous rulings in LBP-00-12 regarding criticality prevention and quality assurance issues." On June 6, 2000, the NRC and CP&L responded to BCOC's petition (ref. 41, 42). In a June 20, 2000, Memorandum and Order (ref. 43), the Commission dismissed BCOC's petition, without prejudice, on the ground that it was prematurely filed. The Commission stated that "After the Board ultimately rules on Orange County's environmental contentions and issues a final decision, Orange County may then resubmit to us its arguments that the Board erred in rejecting the merits of the two contentions concerning criticality prevention and quality assurance."

2.0 BACKGROUND

HNP was originally planned as a four nuclear unit site. In order to accommodate four units at HNP, the fuel handling building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. The two pools at the south end of the FHB, now known as SFPs A and B, were to support HNP Units 1 and 4. The two pools at the north end of the building were to support HNP Units 2 and 3. The multi-unit design included an SFP cooling and cleanup system to service SFPs A and B and a separate cooling and cleanup system to support SFPs C and D.

HNP Units 3 and 4 were canceled in late 1981, and HNP Unit 2 was canceled in late 1983. The FHB, all four SFPs (including liners), and the cooling and cleanup system to support SFPs A

and B were completed. However, the construction on the SFP cooling and cleanup system for SFPs C and D was not completed.

The staff's Safety Evaluation Report, NUREG-1038, "Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," issued in November 1983 (ref. 12) (prior to the cancellation of HNP Unit 2), evaluates the design of all four pools SFPs and the associated cooling systems. The SFP cooling system was designed to consist of one cooling system for each unit. Each cooling system was designed to have two trains of cooling, with each train consisting of a heat exchanger, strainer, and cooling pump. The cooling for the SFP heat exchangers was to be provided by the CCW system for the respective unit. The pools are designed to store both PWR and BWR fuel.

As permitted by its Operating License, CP&L has implemented a spent fuel shipping program. Spent fuel from Brunswick (2 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 will be necessary to activate SFPs C and D and the associated cooling and cleanup systems. Activation of these two pools will provide storage capacity for all four CP&L nuclear units through the expiration of their current licenses.

3.0 EVALUATION

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

a. A revision to TS 5.6 to identify PWR fuel burnup restrictions, BWR fuel 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 is proposing 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 fuel burnup and BWR fuel enrichment to ensure Keff less than or equal to 0.95. The proposed change will involve the addition of Region 2 (non-flux trap style) racks in SFPs C and D in incremental phases (campaigns), on an as-needed basis. In a fully implemented storage configuration, this modification will allow 927 PWR and 2763 BWR fuel assemblies in SFP C. Expansion of storage capacity in SFP D will occur once SFP C is substantially filled. SFP D can accommodate a maximum of 1025 PWR fuel assemblies.

Additionally, in its July 19, 2000, submittal, the licensee submitted updated TS pages to: (1) insert TS 5.6.1.a.2 (renumbered as TS 5.6.1.1.b), which had been inadvertently deleted from the pages submitted on December 23, 1998; (2) revise TS 5.3.1 to reflect the renumbering of TS 5.6.1.a.2; and (3) add Figure 5.6.1 to the TS Table of Contents.

b. 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 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 CCW system 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 American Society of Mechanical Engineers (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.

c. A USQ for additional heat load on the CCW system.

As part of its preparation of the design package for the tie-in of the existing CCW system to provide cooling for SFPs C and D, CP&L prepared a 10 CFR 50.59 evaluation. The scope of the evaluation addressed the tie-in of the Unit 1 CCW system to the heat exchangers of the SFP C and D fuel pool cooling and cleanup system. The evaluation considered a heat load of no more than 1.0 MBtu/hr (consistent with proposed TS 5.6) in SFPs C and D. A thermal-hydraulic model was created to analyze the overall impact of this additional heat load, including its impact on the emergency service water (ESW) system and ultimate heat sink.

A reduction in the minimum specified residual heat removal (RHR) heat exchangers CCW flow from 5600 gpm to 5200 gpm and an increase in the minimum specified CCW heat exchanger ESW system flow from 8250 gpm to 8500 gpm was prescribed by the new thermal-hydraulic analysis in order to maintain all thermal/hydraulic assumptions which are used in the HNP containment analysis. The licensee verified that the minimum specified ESW system flow of 8500 gpm to the CCW heat exchangers was within the capacity of the current system even considering the most limiting ESW system failure. Since the 5600 gpm RHR flow is discussed in NUREG-1038, the licensee determined that the reduction in flow to 5200 gpm was a reduction in an acceptance limit and, therefore, required NRC review.

The staff's evaluation of the licensee's proposed amendment follows.

3.1 Criticality

The licensee requested changes to TS 5.6, "Fuel Storage," to reflect the installation of additional spent fuel storage racks in pools C and D. Pools A and B have already been racked and are nearly full. Pool A contains six Region 1 type (6 x 10 cell) PWR racks and three (11 x 11 cell) BWR racks for a total storage capacity of 723 assemblies. Pool B contains six (7 x 10 cell), five (6 x 10 cell), and one (6 x 8 cell) PWR Region 1 type racks. Pool B also currently contains 17 (11 x 11 cell) BWR racks, 12 of which have been supplied by Holtec International (Holtec). Pool B is licensed to store one more (11 x 11 cell) Holtec BWR rack, which would increase the total pool storage capacity to 2946 assemblies. However, HNP is postponing installation of the last BWR rack in order to reserve the pool open area for fuel examination and repair. Therefore, the combined pool A and B storage capacity will remain as 768 PWR cells and 2057 BWR cells for a total of 2825 storage cell locations.

Under the proposed capacity expansion, storage racks would be added to the unused pools C and D on an as-needed basis. Pool C would provide storage for up to 927 PWR assemblies and 2763 BWR assemblies. Pool D would contain only PWR fuel assemblies with a maximum capacity of 1025 assemblies.

The proposed expansion would consist of the installation of maximum density spent fuel storage racks for pools C and D in phases on an as-needed basis. The new racks were designed by Holtec and are free-standing and self-supporting. The principal construction material is stainless steel. The only non-stainless material is the neutron absorber material, which is a boron carbide and aluminum composite sandwich commonly called boral. Pool C is designed to contain a combination of PWR and BWR assemblies. Pool C can contain two (11 x 9 cell) and nine (9 x 9 cell) PWR racks for storage of 927 PWR assemblies. Pool C can contain two (8 x 13 cell), two (8 x 11 cell), six (13 x 11 cell), and nine (13 x 13 cell) BWR racks for storage of 2763 BWR assemblies. The (9 x 9 cell) PWR racks and the (13 x 13 cell) BWR racks are dimensioned to allow interchangeability between PWR and BWR storage rack styles as required. The racks in pool C will be installed as needed. Pool D contains a variable number of PWR storage spaces. These racks will be installed as needed. Pool D is designed for a maximum storage capacity of 1025 PWR assemblies.

The primary analysis of the reactivity effects of fuel storage in the HNP racks was performed with the CASMO-3 two-dimensional transport theory code. CASMO-3 was also used for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. The MCNP-4A Monte Carlo code was used 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. MCNP-4A was also used for independent verification calculations against CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the HNP spent fuel racks as realistically as possible with respect to important parameters such as enrichment, assembly spacing, and absorber thickness. In addition, these two independent methods of analysis (MCNP-4A and CASMO-3) showed very good agreement with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the HNP storage racks with a high degree of confidence.

The NRC acceptance criterion for subcriticality is that the effective multiplication factor (k-eff) in the spent fuel pool storage racks when fully flooded by unborated water shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95) under all conditions. The criticality analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

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

(2) Unborated pool water at the temperature yielding the highest reactivity (4°C) over the expected range of water temperatures.

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

(4) Neutron absorption in minor structural material is neglected (i.e., spacer grids are analytically replaced by water).

The staff concludes that appropriately conservative assumptions were made.

The Westinghouse 17x17 Standard, 17x17 Vantage 5, 15x15, and the Siemens 17x17 and 15x15 fuel assemblies were evaluated. The design basis fuel assembly used for the PWR rack criticality analyses was the Westinghouse 15 x 15 assembly with a maximum enrichment of 5.0 weight percent (w/o) U-235 since this was determined to have the highest reactivity at zero burnup as a function of burnup for an initial 5.0 w/o U-235 enrichment. For the nominal storage cell design, uncertainties due to manufacturing tolerances on boron loading, boral width, cell inner dimension, stainless steel thickness, and fuel density were included. In addition, a calculational uncertainty for burnup calculations and the effect of axial burnup distribution was included for burnup calculations. These uncertainties were appropriately determined at the 95/95 level, thus meeting the NRC acceptance criterion.

In order to store fuel with maximum initial enrichments up to 5.0 w/o U-235 in the pool C or D PWR racks, the concept of burnup reactivity equivalencing was used. This concept is based on the reactivity decrease associated with fuel depletion and has been previously found acceptable by the NRC for use in PWR fuel storage analyses. A series of reactivity calculations is performed to generate a set of enrichment versus burnup ordered pairs which yield an equivalent k-eff less than 0.95 for fuel stored in the HNP storage racks. The results are illustrated in TS Figure 5.6.1, which indicate that fuel initially enriched to 5.0 w/o U-235 must achieve a burnup of at least 41,447 MWD/MTU to be allowed storage in the PWR spent fuel racks in pools C and D.

The design basis fuel assemblies evaluated for the BWR rack criticality analyses were the General Electric GE-3, GE-4, GE-7, GE-8, GE-9, GE-10, and GE-13 BWR assemblies with a maximum planar average enrichment of 4.6 w/o U-235. The maximum planar average enrichment was assumed for all fuel rods in the assembly. The analyses included the same manufacturing tolerance uncertainties mentioned above for the PWR fuel. These uncertainties were appropriately determined at 95/95 level. In addition, the reactivity increase due to zirconium flow channel bulging was included as well as a reactivity uncertainty in the depletion calculations and an allowance for possible differences between fuel vendor calculations and those reported here.

The analysis for each BWR fuel assembly was performed at the maximum reactivity over burnup. At this point, the assembly was analytically transferred into the storage rack at a reference temperature of 4°C and its k-eff in the rack geometry was determined. The same assembly was also analytically transferred into the HNP standard cold core geometry configuration, which is an infinite lattice with 6-inch spacing at a temperature of 20°C without burnable absorber or control rods and no voids, and its k-inf in this cold core configuration was determined. All xenon which was present during the depletion calculations was removed for the rack and cold core analyses. The maximum calculated CASMO-3 reactivity for each BWR assembly resulted in a k-eff less than 0.95 for the proposed pool C storage racks. This meets the staff's criterion of k-eff no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

Based on these results, a BWR fuel assembly is acceptable for storage in the HNP storage racks in pool C if it has a k-inf in the standard cold core geometry, calculated at the maximum reactivity over burnup, of less than or equal to 1.32. This requirement will be incorporated into HNP TS 5.6.1.

Most abnormal storage conditions will not result in an increase in the k-eff of the racks. However, it is possible to postulate events, such as the accidental insertion of an assembly outside and adjacent to the fuel storage rack, dropping an assembly on top of the rack, lateral rack movement during a seismic event, or the inadvertent placement of a fresh PWR assembly into a location restricted to a burned assembly as per TS Figure 5.6.1, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of soluble boron in the pool water based upon the double contingency principle which requires at least two unlikely, independent, concurrent events to occur before a nuclear criticality accident is possible. Therefore, since soluble boron is normally present in the SFP water, credit for soluble boron may be assumed in evaluating other accident conditions such as the misloading of fresh fuel. Plant procedure CRC-001 requires that the soluble boron concentration in the pool be maintained between 2000 and 2600 ppm and is confirmed by monthly surveillance measurements. The reduction in k-eff due to the boron more than offsets the reactivity addition caused by credible accidents. In fact, Holtec has determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain k-eff less than 0.95 even if a fresh PWR assembly were inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1.

3.1.1 Summary

Based on the review described above, the staff finds that the criticality aspects of the proposed storage capacity expansion for HNP spent fuel pools C and D are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. Our review has also determined that the proposed changes to TS 5.6, "Fuel Storage," are acceptable. In addition, the staff finds the proposed change to TS 5.3, to reflect the renumbering of TS 5.6.1.a.2 to TS 5.6.1.1.b, and the revision to the Table of Contents to reflect the addition of Figure 5.6.1, acceptable.

3.2 Plant Systems

This portion of the evaluation addresses the heat load limitations associated with the expanded capacity for pools C and D, and the resolution of the USQ regarding changes in CCW flow.

3.2.1 Systems Descriptions

HNP was originally planned as a four nuclear unit site. A single FHB was designed and constructed with four separate pools capable of storing spent fuel for all planned units. The two pools at the south end of the FHB were designated A and B, and were designed to store fuel for Units 1 and 4. Pool A, the smaller of the two pools, would store new and spent fuel, while pool B would store spent fuel. The two pools at the north end of the FHB are designated C and D and were designed to support Units 2 and 3. The multi-unit design of the FHB includes a spent fuel pool cooling and cleanup system for pools A and B, and a separate cooling and cleanup system for pools C and D.

Upon cancellation of Units 2, 3, and 4, the licensee decided not to complete the fuel pool cooling system for pools C and D (the cooling and cleanup system for SFPs C and D was approximately 80% complete when construction on the system was halted upon the cancellation of Unit 2). However, the FHB and SFPs A, B, C, and D, including the pool liners, were constructed and turned over to the operating staff as part of the construction and licensing of HNP, Unit 1. The licensee decided not to complete the cooling system for SFPs C and D until these pools were needed for spent fuel storage. The pools have been filled with coolant, but have not stored spent fuel assemblies since they were constructed.

SFP A contains six flux trap style PWR racks and three BWR racks for a total storage capacity of 723 fuel assemblies (FAs). SFP B contains 12 PWR racks and 17 BWR racks, and is licensed to hold one additional BWR rack for a total capacity of 2946 FAs. The licensee proposed the following fuel storage capacities for the four pools (note: the authorized capacity of SFPs A and B will not change):

Pool Designation	Capacity		
A	PWR FA:360	BWR FA: 363	Total: 723
B	PWR FA:768	BWR FA: 2178	Total: 2946
C	PWR FA:927	BWR FA: 2763	Total: 3690
D	PWR FA:1025	BWR FA: 0	Total 1025

Each fuel pool cooling and cleanup system (FPCCS), north and south, is designed with two 100% capacity cooling trains, and a cleanup loop to remove dissolved fission and corrosion products. Each cooling system is comprised of two shell and straight tube heat exchangers, two horizontal centrifugal pumps, a demineralizer, two filters, skimmers, fuel pool and refueling water purification pumps, isolation gates for each pool and transfer canal, and the requisite piping, valves, and system instrumentation. Electrical power for the FPCCS pumps can be aligned to independent emergency supplies. Although originally designed to be cooled by separate cooling water systems, the south and north FPCCS heat exchangers will be cooled by the Unit 1 CCW system. Each pool is outfitted with direct reading temperature and level instruments that provide operators with indication and alarm at local and remote (i.e., the control room) stations.

The CCW system serves as an intermediate closed cooling water system between the radioactive and potentially radioactive systems and the non-radioactive service water system. The FPCCS rejects its heat to the CCW system, which in turn rejects its heat via the service water system to the ultimate heat sink. In addition to the FPCCS, the CCW system provides cooling to various safety-related and non-safety-related heat loads supporting the operation of the reactor. Although the Unit 1 CCW system was originally designed to remove the heat rejected from the spent fuel stored in pools A and B, the system is being modified to remove decay heat from pools C and D as well. The CCW system contains two separate trains, each train containing a CCW heat exchanger. Three CCW pumps are shared by the two trains. During normal and accident operation, including refueling operations, only one CCW pump is

required to be operated to remove the required heat loads from the plant. During plant cool down when heat removal demands on the CCW system are unusually high, two CCW pumps are operated.

3.2.2 Review

The focus of this review is to evaluate the licensee's plans to expand the storage capacity of fuel onsite and the effect the expanded capacity has on the heat removal capabilities of the FPCCSs to ensure they continue to meet staff guidelines on fuel storage. The staff organized the review into four sections:

- A review of the changes proposed to the FPCCS since the original design of the plant was accepted by the staff in NUREG-1038 (ref. 12).
- A review of the effects of the increased decay heat loads on the cooling water systems supporting the storage of spent fuel.
- A review of the heavy loads aspects of this amendment.
- A review of a USQ associated with a proposed reduction of CCW flow to the RHR heat exchangers under certain operating conditions.

The staff based its findings on information contained in the HNP Final Safety Analysis Report (FSAR, ref. 13), NUREG-1038 (ref. 12), and on information contained in letters from the licensee dated December 23, 1998 (ref. 1), and September 3, 1999 (ref. 7).

3.2.2.1 System Design Changes

The staff reviewed and accepted the design of the spent fuel storage system for HNP, Units 1 and 2, in NUREG-1038 (ref. 12). Although the system was never completed, the design of the system was reviewed by the staff in accordance with NUREG-0800, "Standard Review Plan" (SRP, ref. 14). The licensee's amendment dated December 23, 1998 (ref. 1) requested the activation of pools C and D, but also made fundamental changes in the design of the system, for example, changing the system that supplies cooling water to the FPCCS for SFPs C and D from Unit 2 CCW to Unit 1 CCW. As a result, the staff requested in a letter dated August 5, 1999, that the licensee address the differences between the system design that was accepted by the staff in NUREG-1038 and the "as-built" system. The licensee responded by letter dated September 3, 1999 (ref. 7).

In its September 3, 1999, response (ref. 7), the licensee provided a matrix which reconciles the differences between the "as-built" fuel storage system and the conclusions drawn by the staff in NUREG-1038 (ref. 12) concerning the original design of the spent fuel storage system. In general, most of the FPCCS supporting pools C and D was built to the design reviewed by the staff in NUREG-1038 (ref. 12). However, some portions of the system design underwent significant design changes. Those portions have been re-evaluated by the staff and the results are summarized in the following paragraphs.

The staff compared the conclusions drawn by the NRC in Section 9.1.2 of NUREG-1038 (ref. 12) about the original fuel storage system design to the changes to the fuel storage system proposed by the licensee in their December 23, 1998, amendment request (ref. 1). The staff performed this review to ensure the proposed changes did not impact on the staff's previous conclusions concerning the acceptability of design of the fuel storage facility.

In NUREG-1038 (ref. 12), the staff documented the acceptability of the spent fuel storage facility and the SFP cooling systems for both Units 1 and 2 in Sections 9.1.2 and 9.1.3. These sections frequently refer to both Unit 1 and Unit 2. Since Unit 2 was not completed, these references are inaccurate. However, the references are editorial in nature and do not affect the staff's previous conclusions about the acceptability of the fuel storage system.

For those portions of the system covered by Section 9.1.2 of NUREG-0800 (ref. 14), specifically, the FHB, the spent fuel storage racks, the SFP area ventilation system, and other portions of the fuel storage system described in Section 9.1.2 of NUREG-1038 (ref. 12), the staff concluded, based on our review, that the proposed changes do not impact the NRC's previous conclusions and are still acceptable in accordance with the guidance in NUREG-0800 (ref. 14), Section 9.1.2, and Regulatory Guide 1.13 (ref. 15).

The staff also compared the conclusions drawn by the NRC in Section 9.1.3 of NUREG-1038 (ref.12) concerning the original FPCCS design to the changes to the FPCCS proposed by the licensee in their December 23, 1998, amendment request (ref. 1). The staff performed this review to ensure the proposed changes did not impact on the staff's previous conclusions concerning the acceptability of design of the fuel pool cooling and cleanup system.

The licensee called out the following differences between the original FPCCS design accepted by the staff in NUREG-1038 (ref. 12) and the "as-built" system described in the proposed license amendment :

a. A single refueling water storage tank (RWST) to provide system makeup water to both FPCCSs versus an RWST for each cooling system.

b. Emergency makeup for pools C and D provided from the Unit 1 emergency service water (ESW) system, not the Unit 2 ESW system. Flanged connection described in NUREG-1038 (ref. 12) for ESW hookup from Unit 2 will not be installed in the FPCCS system supporting pools C and D. The Unit 1 ESW system is sized to accommodate the emergency fill requirements and can be cross-connected to all pools.

c. The current design limits the temperature of SFPs A and B to 137°F, assuming a single active failure, which is lower than the temperature stated in NUREG-1038 (ref. 12).

d. SFP chemistry limits are currently maintained consistent with guidelines established by the NSSS vendor, fuel manufacturer, and EPRI guidelines. NUREG-1038 (ref. 12) assumed a weekly sampling protocol.

e. NUREG-1038 (ref. 12) contains many references to Unit 2. Due to the cancellation of Unit 2, references to GDC 5, sharing of structures, systems and components, are no longer applicable.

Items c, d, and e, above were reviewed by the staff and found to be editorial in nature, or approved by the staff in previous licensing actions and part of the current system design basis, and are, therefore, acceptable.

For items a and b above, the staff previously accepted a design for the fuel storage system whereby separate RWSTs would be available to provide makeup water to pairs of SFPs (SFPs A and B from Unit 1 RWST, SFPs C and D from Unit 2 RWST). Similarly, the staff accepted a backup method of makeup to the fuel storage system from the Unit 1 and Unit 2 ESW systems through valved and flanged connections. Since Unit 2 was never constructed, the Unit 2 RWST and the Unit 2 ESW system are not available. Makeup from the RWST is used to compensate for coolant losses due to evaporation and cooling system leakage. The proposed change recognizes the Unit 1 RWST as the seismic Category 1 makeup water source for both FPCCSs, supplying makeup for all four SFPs. Similarly, the Unit 1 ESW system is available to provide a seismic Category 1 backup makeup water source through a cross-tie to all four fuel storage pools in the event of an emergency. The licensee has evaluated this configuration and determined that the Unit 1 RWST and the Unit 1 ESW system have sufficient capacity to supply makeup for all four pools. The staff reviewed the proposed changes to the seismic Category 1 makeup supplies for the FPCCS for pools C and D and finds that the Unit 1 RWST and the Unit 1 ESW system have sufficient capacity to provide makeup to the four fuel storage pools and are, therefore, acceptable.

In addition to the changes made to the systems that directly support SFPs C and D, the Unit 1 CCW system was also modified to account for the absence of the Unit 2 CCW system. Valves, piping and other components were added to the Unit 1 CCW system to provide heat removal capability for the SFPs C and D FPCCS heat exchangers. The staff's evaluation of the effects of adding an additional heat load to the Unit 1 CCW system is included in Section 3.2.2.3 of this Safety Evaluation.

3.2.2.2 Changes in Decay Heat Load

The licensee provided a summary of methods, models, analyses, and numerical results to demonstrate the compliance of the HNP spent fuel storage systems with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications" (ref. 16). The licensee provided the following analyses as justification for the acceptability of the proposed changes to the spent fuel storage system:

- Evaluation of the long-term decay heat load in SFPs C and D.
- Evaluation of the steady-state bulk pool temperature with forced cooling available (fuel pool bulk temperature is limited to 137°F with the FPCCS in operation).
- Determination of the maximum pool local temperature.
- Evaluation of the potential for flow bypass from the pool inlet to the pool outlet with the sparger removed.
- Evaluation of the time-to-boil assuming all forced cooling is lost.

Holtec International, a contractor of the licensee, performed decay heat load calculations in accordance with USNRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." (ref. 1, Enclosures 6 & 7). The calculations assumed that the spent fuel stored in SFPs C and D had cooled a minimum of 5 years before being placed in SFPs C or D. Holtec determined the bounding decay heat load in SFPs C and D based on fuel characteristics documented on Tables 5.2.1 and 5.2.2 of Enclosure 6 to the letter dated December 23, 1998 (ref. 1). Although the bounding calculations determined that the maximum decay heat load in SFPs C and D could reach 15.63 Mbtu/hr, the licensee has decided to limit the maximum decay heat load in SFPs C and D to 1 Mbtu/hr using administrative controls.

In an August 5, 1999, Request of Additional Information (RAI), the staff asked the licensee to provide an analysis showing that the maximum bulk temperature for SFPs C and D will not be exceeded assuming an increase in the decay heat load of 1 Mbtu/hr. In a letter dated September 3, 1999 (ref. 7), the licensee provided an analysis that shows the maximum bulk temperature of all four SFPs remains below the pool design temperature of 137°F under a variety of operational conditions, including those that conform to the guideline in NUREG-0800 (ref. 12) for partial and full core offloads. Where appropriate, the licensee assumed a single active failure and design temperatures (e.g., 95°F for the ESW system) in the systems providing cooling water to the FPCCS heat exchangers.

The staff reviewed the documentation and agrees with the licensee that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure.

The licensee's contractor, Holtec, also performed an analysis of the temperatures at various locations in the fuel pool to ensure localized boiling does not occur, especially in the fuel storage racks. Bounding assumptions for fuel storage location and cooling times were assumed, as well as for bulk coolant temperature and cooling flow to the SFPs. A computational fluid dynamics model was used to determine the difference between peak local and bulk coolant temperatures. The results indicate that peak local temperature in the pool will be 6.8°F higher than the maximum bulk coolant temperature of 137°F. Based on a review of the licensee's methods and findings, the staff agrees that sufficient thermal margin exists to preclude localized boiling.

The licensee also provided the results of heat up calculations to determine the time-to-boil should a loss of all forced cooling occur. Section 5.4.1 of Enclosure 6 of the letter dated December 23, 1998 (ref. 1) discusses the results of time-to-boil calculations performed by Holtec. The results indicate that with a heat load of 15.63 Mbtu/hr in SFPs C and D, and an initial bulk coolant starting temperature of 140°F, more than 13 hours are available to take mitigating action. The staff considered this evaluation very conservative, given the heat load in SFPs C and D will be limited to 1 Mbtu/hr, and requested that the licensee evaluate the pool under its expected operating conditions. The licensee performed additional calculations that indicate several hundred hours are available to mitigate a total loss-of-cooling event in SFPs C and D assuming a 1 Mbtu/hr heat load limit. These calculations are documented in a letter dated September 3, 1999 (ref. 7).

The staff performed an independent heat-up evaluation to ensure the licensee's results were conservative. For added conservatism, the staff assumed the SFPs were isolated from each other when cooling was lost and that the entire decay heat load was located in a single pool. The staff's evaluation confirmed that more than 100 hours are available to identify and address a loss of all forced cooling event if the heat load were limited to SFP C, and more than 50 hours are available if the decay heat load were limited to SFP D.

Given the decay load in SFPs C and D will be limited to 1 Mbtu/hr, the staff agrees that sufficient time is available for plant operators to take mitigating actions prior to pool boiling.

3.2.2.3 Unreviewed Safety Question

The CCW system provides cooling to the RHR system heat exchangers, RHR pumps, the SFP heat exchangers, and other non-safety-related systems. Two RHR trains provide long-term cooling during the containment sump recirculation phase of a loss-of-coolant accident (LOCA) by circulating the reactor coolant from the containment sump, through the heat exchangers, and returning it to the reactor coolant system cold legs. Each RHR train is capable of removing up to 111.1 Mbtu/hr in the post-LOCA scenario. In the USQ thermal-hydraulic analysis, the licensee demonstrates that adequate excess thermal capacity existed in the CCW system to accommodate the additional heat loads of 1.0 MBTU/hr (which is a limitation specified in TS 5.6.3) from SFPs C and D during all normal and accident modes of system operation, i.e., the required RHR heat removal capability can be met with reduced CCW flow through the RHR heat exchanger due to the tie-in of the C and D FPCCS.

The USQ thermal-hydraulic calculations did not change any assumptions regarding maximum sump temperatures or RHR heat removal requirements under post-LOCA containment conditions. However, the licensee identified that fluid properties at the higher RHR temperatures associated with the post-LOCA scenario would result in an increase in the heat exchanger heat transfer coefficient values over the fixed value assumed in the existing analysis. The analyses used a "dynamic" RHR heat exchanger performance model in which the tube side inlet temperature is postulated to rise to 244.1°F during the initial phase of containment sump recirculation, rather than a fixed 139°F currently assumed. This increased tube side fluid temperature increases the overall RHR heat exchanger heat transfer coefficient (HTC) by approximately 10% due to the change in tube side fluid viscosity. Based on this increased heat exchanger HTC, the calculations showed that a minimum CCW system flow rate through the RHR heat exchanger of 4874 gpm at 120°F is required at the beginning of the sump recirculation phase. Assuming a 6% model uncertainty, the required CCW system flow to the RHR heat exchanger would be 5166 gpm, which is less than the 5600 gpm required by the existing analysis.

The licensee also provided, in response to a staff question (question 6, September 3, 1999, letter (ref. 7)), the results of analyses based on a time-dependent heat rejection load of the RHR heat exchanger, and the containment sump water temperature during a LOCA. The staff has performed an audit calculation of these results, and found that the analyses were conservatively based on a lower density and mass flow of the CCW volumetric flow rate of 4874 gpm. The staff concurs with the licensee's analysis conclusion that the required RHR heat removal capability can be met with the reduced CCW flow of approximately 5200 gpm.

3.2.2.4 Summary

The licensee proposed to modify Section 5.6.3, "Capacity," of the TS to define the maximum capacity of the four SFPs. In addition, the licensee included a section to limit the total decay heat load in SFPs C and D to 1 Mbtu/hr. The licensee also identified a USQ and provided a justification why the changes to the design of the CCW system were acceptable. Information provided by the licensee in the amendment request dated December 23, 1998 (ref. 1), and letter dated September 3, 1999 (ref. 7), documented the licensee's justification for requesting the staff's approval of this amendment.

The staff has completed its review of the USQ justification and the decay heat load aspects of increasing capacity of the fuel storage system at HNP. Based on the above evaluation, the staff finds the licensee proposed changes acceptable.

3.3 Handling of Heavy Loads and Spent Fuel Assemblies

3.3.1 Background

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. NUREG-0612 provides guidelines that are implemented in two phases. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads and provide criteria for establishing safe load paths, procedures for load handling operations, training of crane operators, design, testing, inspection, and maintenance of cranes and lifting devices, and analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased handling system reliability, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drops and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety.

NRC Safety Evaluation Report related to the operation of HNP, Unit 1, NUREG-1038, Supplement No. 4, dated October 1986 (ref. 12) approved CP&L's NUREG-0612, Phase I heavy loads program. In the proposed amendment, the licensee addresses heavy load issues, including the installation of spent fuel storage racks in SFPs C and D, fuel movement, movement of the gates that isolate the pools from the transfer canal, and movement of spent fuel dry storage casks. Considerations are given to the design and operation of the hoisting

systems, safe load paths, procedures, crane operator training, and postulated load drop accidents and consequences on fuel and on the SFP.

3.3.2 Review

3.3.2.1 Hoisting Systems

The FHB auxiliary crane, which is rated at 10 tons, will be used to lift and move the new racks, the gates that isolate the pools from the transfer canal, and new FAs. The racks will be lifted up through the equipment hatch then transported along the safe load path to SFPs C and D. The same crane will be used to lower the racks into SFPs C and D. A 20-ton hoist will be suspended from the bridge of the FHB auxiliary crane and used in conjunction with the spent fuel rack lifting rig (special lifting device) to lift and move the racks into the pools. The use of the 20-ton hoist will allow the licensee to avoid contamination of the main hook during SFP rack movement in the pools. The 150-ton Spent Fuel Cask Handling Crane will be used to lift and move shipping casks containing offsite spent fuel from the railroad car through the equipment hatch to the cask loading pool.

In NUREG-1038, Supplement No. 4 (ref. 12), the NRC staff approved the load handling systems, including the FHB auxiliary crane and the spent fuel cask handling crane. The staff found that the load handling system provided adequate protection against heavy load drops and was consistent with NUREG-0612. However, the spent fuel storage rack lifting rig is specifically designed to lift the new rack modules; therefore, it was not addressed in NUREG-1038.

Both the FHB Auxiliary Crane and the Spent Fuel Cask Handling Crane are designed, fabricated, installed, inspected, tested, and operated in accordance with requirements of the Crane Manufacturers Association of America (CMAA) Specification No. 70, "Specifications for Electric Overhead Traveling Cranes," and ANSI B30.2-1976, "Safety Standards for Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)." The FHB auxiliary crane is single-failure-proof and although it has a rated capacity of 10 tons in the auxiliary hook, it can be used to handle items that weigh more than 10 tons but less than 12 tons provided that they are evaluated and administratively controlled. The single-failure-proof design of the crane enables the licensee to retain and hold the load in a stable and immobile safe position during an event. The crane is equipped with a means to safely move the load manually to a lay down area for emergency manual lowering of the load. Also, all the components in the load path of the crane hoist such as the hook, hoist rope, reeving, and braking mechanisms, either are redundant or have a large factor of safety. In addition, the crane is designed to maintain its structural integrity and hold the load under the dynamic loading conditions of a safe shutdown earthquake (SSE). The maximum lifted weight during rack installation includes the rack, lifting rig (special lifting device), rigging, and the 20-ton hoist for a total weight of 18,820 lbs.

The licensee states that the rack modules will be lifted using a remotely engaged spent fuel rack lifting rig that is specifically designed to lift both PWR and BWR spent fuel rack modules. The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612, Sections 5.1.6(1) and 5.1.6(3a), and the requirements in ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials." Accordingly, in accordance with NUREG-0612, the lifting rig has twice the design safety factor with respect to the yield and ultimate strength (six (6) times and ten (10) times the

combined concurrent static and dynamic loads for the yield and ultimate strength, respectively) of its material of construction. The lifting rig also is redundantly designed with four independently loaded lift rods that are configured such that failure of a single rod will not result in uncontrolled lowering of the rack. Therefore, the lift rods and lift points of the lifting rig are designed and tested as follows: (1) with a stress design factor of five times the lifted weight without exceeding the ultimate strength of the material; and (2) load tested to 300% of the maximum weight to be lifted, and hoisted and suspended for a minimum of 10 minutes. After load testing, examination of the critical weld joints using a liquid penetrant is performed.

The staff finds that the lifting capacity of the 10-ton single-failure-proof FHB auxiliary crane coupled with the capacity of the 20-ton hoist and the spent fuel rack lifting rig (special lifting device) will support the weight of the racks and the added rigging loads. In addition, the design, inspection and testing of the crane and lifting device will help to assure the licensee's safe handling of the racks with little to no risk of an accidental rack drop during rack installation.

3.3.2.2 Load Path

The new spent fuel storage racks will be installed in five sequential campaigns as follows:

	SFP C (date)	SFP D (date)
Campaign 1	14 (2000)	6 (2016)
Campaign 2	10 (2005)	6 (TBD)
Campaign 3	6 (2014)	

The new racks will be lifted to the FHB operating level through the equipment hatch, then moved along the safe load path that was previously identified as the path used for the spent fuel shipping cask. The racks are then moved over the fuel transfer canal from which they are moved over SFP C and D and lowered into position. The safe load path from the equipment hatch to the SFPs is clear of any safety-related equipment. The licensee stated that rack installation and fuel assembly storage will begin in the south end of SFP C and proceed north to SFP D. Therefore, lifts of the racks over spent fuel will be avoided.

As stated by the licensee, the new installed fuel storage racks will not significantly change the method of handling loads during normal plant operations because the same equipment (i.e., the spent fuel handling machine and tools) and procedures as those currently used in pools A and B will be used in pools C and D.

New and spent fuel shipping casks are lifted from the carrier through the equipment hatch up to the FHB operating level by the FHB auxiliary crane or the spent fuel cask handling crane. As stated in FSAR Section 9.1.4.3.2(b), spent fuel shipping casks are handled by the 150-ton cask handling crane. Therefore, the shipping casks containing offsite spent fuel will be lifted by the cask handling crane through the equipment hatch to the operating level then moved to the cask loading pool. Permanent mechanical stops on the cask handling crane limit any travel of the crane over the SFPs. This enables the licensee to avoid traversing or dropping the cask over spent fuel in the fuel pools.

The staff finds that the load paths for movement of the spent fuel storage racks and the pattern for storing spent fuel in the racks does not involve any movement of the racks over spent fuel. Also, cask movements do not involve travel over fuel stored in the racks or over the SFP.

3.3.2.3 Analysis of Heavy Load Drop Accidents

The licensee analyzed postulated load drops of spent fuel assemblies, spent fuel storage racks, and the gates that isolate the pool from the transfer canal. Two drops of the fuel assembly using a bounding impact weight of 2100 lbs. (includes the heaviest fuel plus the handling tool) was considered: vertical drops on top of the racks, and vertical drops to the base plate of the racks. The FA drop on top of the spent fuel racks resulted in deformation of the racks to a depth of 11 inches below the top of the rack. However, the fuel in the racks would not be damaged. The FA drop onto the base plate of the racks resulted in damage to the baseplate but no damage to the SFP liner.

HNP FSAR (ref. 13), Appendix 9.1A, "Heavy Loads Analysis," does not address the drop of a rack into the SFP. However, because a single-failure-proof crane is used, there is a reasonably low chance of a rack drop. Nonetheless, in its amendment request, the licensee considered a vertical drop of the heaviest rack (16140lbs.) from 40 feet above the SFP floor liner. The results indicated that some damage to the SFP liner and minor damage to the SFP concrete floor slab would occur. In telephone conferences on March 30 and April 4, 2000, the staff discussed additional information it needed on the results of the rack drop analyses. By letter dated April 14, 2000 (ref. 10), the licensee responded that a rack drop would pierce the SFP liner and result in leakage of SFP water. However, the plant's design basis leakage detection system is designed and operated to detect, limit, and contain leakage from the SFP. Valves in the leakage detection system are normally closed and are only opened to check for and measure any leakage during the operator monthly rounds. Therefore, the closed valves will enable the system to limit any SFP leakage. In addition, SFP makeup can be made available from a number of sources to supplement any leakage from the SFP. Emergency makeup can be provided from the emergency service water system. Normal SFP makeup can be provided from the demineralized water system, RWST, the reactor coolant drain tank, and the reactor makeup water storage tank. Therefore, due to these capabilities, and because the structural integrity of the concrete slab remained unimpaired after the drop, the licensee concluded that neither catastrophic damage of the SFP structure nor rapid loss of pool water would occur.

The licensee did not analyze the potential for a rack drop on spent fuel assemblies or on safety-related equipment because: (1) the racks would not be moved directly over any fuel in the pool; and (2) upending and laying down the racks is planned to occur in an area that avoids any potential impact on safety-related equipment.

The drop of the gates (4,000 lbs. each) that isolate the pool from the transfer canal was analyzed. The gates will also be lifted using the single-failure-proof auxiliary crane and dual rigging that satisfies NUREG-0612 safety margins. The gate rigging will be load-tested to lift three times the weight of the rack and the other components of the lifting device without exceeding the minimum yield strength of the material. The gate rigging also will be capable of lifting five times that weight without exceeding the ultimate strength of the rig materials. The postulated gate drops were analyzed at 15 inches above loaded fuel racks and at 40 feet above the SFP liner. A gate drop would penetrate the racks to a depth of 5 inches with no impact on

the stored fuel. It also would damage the pool liner; however, the SFP concrete slab would not fail. Therefore, fuel in the racks would not be affected if a gate drop occurred. If the SFP liner is breached, the leakage detection system and makeup capability as discussed above would be effective.

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using administrative controls and procedures to preclude load drop accidents in these four areas. Accordingly, the licensee plans to provide: (1) comprehensive training to the rack installation crew in accordance with ANSI B30.2; (2) redundantly designed and adequately tested lifting rigs in accordance with ANSI N14.6; (3) inspection and maintenance checks on the cranes, lifting devices, and racks themselves prior to and during the rack installation; and (4) specific procedures that cover the entire racking effort, including the identification of required equipment, inspection, acceptance criteria prior to load movement, defining safe load paths, and steps and precautions for proper load handling and movement. In addition, the licensee states that its rack installation pattern will enable the racks to be lifted and inserted without travel over spent fuel in the SFP during the rack installation operation.

Since both the spent fuel cask handling crane and the configuration of the FHB are designed to avoid any travel of the crane hook over the SFPs, the spent fuel shipping cask will not be moved over or have any opportunity to fall into the fuel pools. As a result, there is no need for a load drop analysis of the cask over the SFPs.

The staff accepts the licensee's finding that, based on the load drop analyses, the integrity of the fuel and the SFP would be maintained if an FA or a spent fuel storage rack is dropped. The use of a single-failure-proof crane in conjunction with administrative procedures and controls that are focused on, but not limited to, the areas noted above would enable the licensee to maintain safety during the implementation of the proposed changes.

3.3.2.4 Summary

Based on the preceding discussions, the staff finds that the aforementioned considerations for the movement of heavy loads to support the proposed changes to TS 5.6 and the increase in the SFP storage capacity are acceptable. The licensee's use of the 10-ton single-failure-proof FHB auxiliary crane, the 20-ton hoist, the spent fuel rack lifting rig, and administrative controls and procedures that are in accordance with NUREG-0612 and ANSI N14.6, will help to maintain safety during the installation of the new racks. The reliability of the crane coupled with the design, testing and inspection of the crane, the lifting rig and other lifting devices will enable the licensee to handle safely the racks and other heavy loads during the rack installation process. The postulated accident analyses involving a dropped spent fuel storage rack and the gate indicated that the SFP liner could be breached. However, during such a breach, the licensee could maintain the pool and its contents within the acceptable consequence limits set forth in NUREG-0612. In addition, the licensee's use of administrative controls and procedures to improve the handling and control of heavy loads, including the racks, enhances the licensee's capability to reduce the potential for load drops.

In addition, the licensee's cask handling operations will not occur over the SFPs. Therefore, movement of the cask will not impact the new racks nor the fuel configuration in the pools. Also, the licensee stated that there will be no significant changes in its method of handling new and spent fuel assemblies subsequent to the installation of the racks. Therefore, the staff believes that the existing fuel handling procedures as documented in the FSAR (ref. 13) will enable the licensee to continue to maintain safety during its fuel handling operations.

3.4 Structural Engineering

The primary purpose of this review is to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structures subject to the effects of the postulated loads (Appendix D of SRP Section 3.8.4)(ref. 14) and fuel handling accidents.

3.4.1 Storage Racks

CP&L has proposed to install 30 spent fuel storage racks in SFP C, and 12 racks in SFP D. The total storage capacity of the 30 racks is 3690 storage locations and the total storage capacity of the 12 racks is 1025 storage locations. All 42 storage racks are seismic Category I equipment and are required to remain functional during and after an SSE. CP&L, with its contractor Holtec, performed structural analyses of the racks for the requested license amendment.

The computer program DYNARACK was used for dynamic analysis to demonstrate the structural adequacy of the HNP spent fuel rack design under the combined effects of earthquakes and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment, and they are not attached to the floor or walls of the SFP. A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements and friction elements, as defined in the program, were used to simulate the three dimensional (3-D) dynamic behavior of the rack and the stored FAs, including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Analyses of two models were performed: a 3-D single-rack (SR) model and a 3-D whole pool multi-rack (MR) model. For the 3-D MR analyses, all racks were considered to be fully loaded and partially loaded with three different coefficients of friction (μ=0.2, 0.8 and a random value where the mean is about 0.5) between the rack pedestal and the pool floor to investigate the fluid-structure interaction effects between the racks and the pool walls as well as those among the racks and to identify the worst-case response for rack movement and for rack member stresses. For the 3-D SR analyses, the rack was considered to be half loaded with a condition that all loaded fuels were placed at one side of the rack. The coefficient of friction of 0.8 between the rack pedestal and the pool floor was used to investigate the stability of the rack with respect to overturning.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) were generated from the design response spectra defined in the FSAR (ref. 13). CP&L demonstrated the adequacy of the single artificial time history set used for the seismic analyses

by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density (PSD) function compatible with the design response spectra as discussed in SRP Section 3.7.1 (ref. 14).

A total of 17 3-D SR and MR analyses were performed. The racks were subjected to the service, upset and faulted loading conditions (Level A, B and D service limits). The results of the analyses show that the maximum displacement of the racks at the top is about 1.494 inches, indicating that there is adequate safety margin against overturning of the racks. The results of the analyses also show that there is no impact potential between the rack and the pool wall. However, the results show that there is impact potential between the racks. The staff compared the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF. The stress results show that the induced impact forces under the SSE loading condition are small and all induced stresses in the racks are smaller than the corresponding allowable stresses specified in the ASME B&PV Code, indicating that the rack design is adequate.

CP&L also calculated the rack weld stresses at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal and cell-to-cell connections) under the dynamic loading conditions. CP&L demonstrated that all of the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on (1) CP&L's comprehensive parametric study (e.g., varying coefficients of friction and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when they are compared to the corresponding allowables provided in the ASME B&PV Code, and (3) CP&L's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

CP&L analyzed the SFPs to demonstrate the adequacy of the structures under fully loaded fuel racks with all storage locations occupied by FAs. The fully loaded structures were subjected to the load combinations specified in the HNP FSAR (ref. 13).

The licensee's July 23, 1999, submittal (ref. 6) shows the predicted factors of safety varying from 1.05 to 3.51 for shear force and bending moment of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the CP&L structural analyses demonstrate the adequacy and integrity of the structures under full fuel loading, thermal loading and SSE loading conditions. Thus, the storage fuel pool design is acceptable.

3.4.3 Fuel Handling Accident

The following two refueling accident cases were evaluated by CP&L: (1) drop of an FA with its handling tool, which impacts the baseplate (deep drop scenario); and (2) drop of an FA with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis results of accident case (1) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the FA drop through the rack structure. The analysis results of accident drop case (2) show that damage will be restricted to a depth of 11.0 inches below the top of the rack, which is above the active fuel region. The staff reviewed CP&L's analysis results and concurs with its findings. These results are acceptable based on CP&L's structural integrity conclusions supported by the parametric studies.

3.4.4 Summary

Based on the review and evaluation of CP&L's submittals, the staff concludes that CP&L's structural analysis and design of the spent fuel rack modules is acceptable and that the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14), and are, therefore, acceptable.

3.5 Materials Engineering

The new maximum storage rack arrays proposed for use in the SFP are manufactured by Holtec. The racks are free-standing and self-supporting. The racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME B&PV Code.

3.5.1 Structural Materials

The structural materials used in the fabrication of the new spent fuel racks include: (1) ASME SA240-304L for all sheet metal stock and internally threaded support legs; (2) ASME SA564-630 precipitation hardened stainless steel (heat treated to 1100°F) for externally threaded support spindle; and (3) ASME specification SFA 5.9 ER308L for weld material.

These materials used in the Holtec racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the SFP environment. Therefore, they are acceptable for use in this application.

3.5.2 Poison Material

The Holtec racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the SFP environments, where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed to vent the corrosion gases. The neutron-absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

3.5.3 Summary

Based on the above evaluation, the staff finds that the materials utilized in the fabrication of the spent fuel racks manufactured by Holtec are compatible with the SFP environment at HNP. The type of degradation exhibited by the racks does not affect their neutron-absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.6 Radiological Assessment

3.6.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for placing SFPs C and D in service with respect to occupational radiation exposure. For this modification, the licensee plans to install region 2 (non-flux trap style) rack modules in pools C and D in incremental phases, on an as-needed basis. The licensee estimates that the collective dose associated with the proposed fuel rack installation is in the range of 2-3 person-rem.

All of the operations involved in racking will utilize detailed procedures prepared with full consideration of ALARA (as low as reasonably achievable) principles. The HNP racking project represents lower radiological risk due to the fact that the pools currently contain no spent fuel. The Licensee's Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels, and dosimetry requirements. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment, including alarming dosimeters.

Since this license amendment does not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to utilize divers to remove any interferences that may impede the installation of the new spent fuel racks, the licensee will equip each diver with the appropriate monitoring equipment. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposure is maintained ALARA.

On the basis of our review of the licensee's proposal, the staff concludes that the expansion of HNP's spent fuel storage capacity can be performed in a manner that will ensure that doses to workers will be maintained ALARA. Therefore, the staff finds the licensee's proposal acceptable.

3.6.2 Solid Radioactive Waste

The necessity for pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally expected to be changed about once a year. The licensee does not expect the resin change-out frequency of the SFP purification system to be permanently increased as a result of the expanded storage capacity. Overall, the licensee does not expect that the additional fuel storage made available by the increased storage capacity will result in a significant change on the generation of solid radioactive waste. The staff finds the licensee's conclusion to be acceptable.

3.6.3 Design Basis Accidents

In its application, the licensee stated that since the pertinent fuel parameters remain unchanged, the radiological dose consequences at the exclusion area boundary for the accidental drop of a fuel assembly in the SFP will not be increased from those previously calculated. Section 15.7.4.5 of the HNP FSAR (ref. 13) describes the worst-case fuel handling accident in the FHB for dose consequences. This case assumes a PWR assembly is dropped, initially strikes a stationary PWR spent fuel assembly, then falls onto an adjacent loaded BWR spent fuel rack. This scenario assumes 314 PWR spent fuel rods and 52 BWR spent fuel assemblies fail. FSAR Table 15.7.4-7 (ref. 13) gives the offsite dose consequences for the above bounding scenario. The addition of racks to the currently unused SFPs C and D would not change the bounding dose scenario. No change is being made to the handling of the spent fuel or the types of fuel stored in the HNP SFPs.

The staff agrees that the bounding scenario for the postulated fuel handling accident in the FHB does not change due to the addition of storage racks in SFPs C and D. Therefore, the inputs and assumptions for the dose consequences analysis do not change, and the current fuel handling accident dose assessments in the HNP FSAR (ref. 13) remain bounding. The staff agrees with the licensee's assessment and finds their conclusion acceptable. The staff has determined that the addition of spent fuel racks to SFPs C and D is acceptable with regard to the radiological consequences of the postulated fuel handling accident.

3.7 10 CFR 50.55a Alternative Plan

As part of its December 23, 1998, application (ref. 1), the licensee proposed an alternative to the ASME B&PV Code (Code) documentation requirements for certain pipe welds in the cooling and cleanup water systems servicing SFPs C and D. The alternative was necessitated by an HNP determination that some documents required by the Code for welds were destroyed. The alternative consists of varying degrees of inspection coverage performed by a variety of examination methods, selective nondestructive testing (NDE), selective weld replacements, selected chemical analyses, and establishing that the missing documents had existed and had been found acceptable by the responsible parties prior to their destruction.

In a March 24, 1999, RAI, the NRC requested clarification of certain aspects of the licensee's December 23, 1998, letter (ref. 1). The licensee responded to the RAI in a submittal dated April 30, 1999 (ref. 4). The staff sent the licensee a second RAI on September 20, 1999. The licensee responded in submittals dated October 15, 1999 (ref. 8), and October 29, 1999 (ref. 9). Additional information was gathered during an NRC staff on-site inspection of the SFPs on November 15 through 19, 1999 (ref. 17). A follow-up inspection was conducted from January 31 through February 4, 2000 (ref. 19).

3.7.1 Background

The licensee filed an application with the Atomic Energy Commission (AEC) in 1971 for licenses to construct and operate HNP Units 1, 2, 3, and 4. The AEC issued construction permits for each of the four units in January 1978. Construction proceeded until December 1981, when the licensee informed the NRC that Units 3 and 4 were being canceled, and requested that Units 1 and 2 be considered concurrently for operating licenses. NUREG-1038 (ref. 12) was issued in

November 1983 for Unit 1, and reflected ongoing construction for Unit 2. In December 1983, Unit 2 was canceled. Unit 1 was issued an operating license in January 1987.

The FHB was designed to have four SFPs. SFPs A & B were to support Units 1 and 4 and SFPs C & D were to support Units 2 and 3. These concrete pools are interconnected, which dictated that they be completed at least to the point of being watertight. For a watertight SFP, the reinforced concrete encasement, stainless steel pool liners, and closure of certain pipe openings had to be in place. By the time Units 2, 3, and 4 were canceled, the majority of mechanical piping and equipment associated with the operation of the pools were already installed. The four SFPs, the FPCCS for SFPs A & B, and the CCWS for SFPs A & B were completed and turned over as part of the construction and licensing of Unit 1. The construction of the FPCCS and CCWS for SFPs C & D was discontinued and their pipe stubs extending outside of the pools were capped. Some of the field weld documentation associated with the FPCCS and CCWS for SFPs C & D were inadvertently discarded during subsequent document control cleanup efforts.

10 CFR 50.55a(g)(4) states that ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the Code and addenda that become effective subsequent to the editions specified in paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

10 CFR 50.55a(a)(3) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.7.2 Requirements

The licensee filed an application with the AEC in 1971 for licenses to construct and operate HNP, Units 1, 2, 3, and 4. The construction code used to design, fabricate, erect, construct, and inspect the balance of plant piping requiring a Code stamp was the ASME Code, Section III, Division 1, 1971 Edition with Summer 1973 Addenda. Field fabrication and installation of Code items was performed in accordance with the ASME Code, Section III, Division 1, 1974 Edition with Winter of 1976 Addenda.

The specific paragraphs in the Code that pertain to missing documentation as discussed in this Safety Evaluation are listed in the licensee's letter dated April 30. 1999, Enclosure 7, "Matrix of Code Requirements vs. Missing Field Records" (ref. 4).

3.7.3 Licensee's Proposed Alternative

For the missing SFPs C & D construction documentation, the licensee's proposed alternative is to reinspect and reconstitute the records to the 1974 Edition with 1976 Winter Addenda of Section III of ASME Code. For the items that cannot be reinspected or reconstituted to Code,

the proposed alternative is to perform chemical analyses/checks, visual examinations, and record reviews in order to establish an acceptable level of quality and safety.

The proposed alternative for specific paragraphs in the Code that pertain to missing documentation are listed in the licensee's letter dated April 30. 1999, Enclosure 7, "Matrix of Code Requirements vs. Missing Field Records" (ref. 4).

3.7.4 Basis for the Alternative

The licensee developed the proposed alternative based on the accessibility of the welds. For field welds that were accessible, the licensee relied mostly on re-inspections and testing. For inaccessible embedded field weld, the licensee relied mostly on non-Code methods of inspections, testing, and alternate documentation.

The weld filler material used for all accessible stainless steel field welds in the FPCCS piping were subjected to a limited chemical analysis with an "alloy analyzer." In addition, chip samples from three randomly selected field welds were subjected to complete chemical analyses. The chemical analyses demonstrate consistency and supported the validity of the chemical results from the alloy analyzer. Also, a limited number of these field welds were checked for ferrite number. The ferrite number checks provided supporting data of the acceptability of the stainless steel filler metal used in these applications.

All three CCWS carbon steel field welds with missing documentation were analyzed for material composition. The analyses verified the acceptability of filler metal for the application.

The as-built isometric drawings were reconstructed from data gathered during a detailed walk-down inspection of the current configuration.

The personnel used to complete and license SFPs A & B came from the same labor pool and used the same procedures as were used on the completed portions of SFPs C & D and their support facilities.

The NDE for the welds with missing documentation was recorded on the weld data record (WDR). WDRs are the only licensee records attesting to the Code acceptability of NDE. HNP recreated, to the extent possible, new WDRs for each field weld within the scope of the alternative. The recreation consisted of NDE reinspection of the accessible welds to the original construction Code. For field welds embedded in concrete, HNP visually examined them from inside the pipe. The examinations provided subjective information on the weld quality and, to the extent feasible, objective evidence of compliance with Code and procedural requirements.

The licensee used the existence of hydrostatic test records as evidence that WDRs were completed and contained the appropriate NDE records. The reinspections of accessible field welds with no rejections provided supporting evidence of acceptability of the construction process. The hydrostatic test records and reinspection records assure that missing WDRs had been satisfactorily completed with acceptable NDE results.

The initial construction program used the same pool of qualified craftsmen, quality control personnel, and engineers for all four SFPs. The construction program also used the same type of material, the same procedures, and the same program controls. The adequacy of these site programs and controls is validated by the acceptability of the operational SFPs A & B, including the acceptability of the construction documentation for SFPs A & B.

Beyond programmatic assurances, a large body of evidence was compiled attesting to the quality of construction of SFPs C & D. Vendor data packages, hydrostatic test records, quality control records, and other construction-era documentation were retrieved that validate compliance with site programs and procedures. An examination effort was completed in which Code-required external NDE of accessible welds were reexamined with no rejectable indications and filler metals were examined for chemical composition verifying material selection. These results provide direct evidence of the quality of accessible field welds, and by extension, the smaller group of welds that are embedded.

In the absence of any contradictory evidence, and the preponderance of supporting evidence, the licensee concludes that the construction that occurred during the original construction phase of SFPs C & D was performed to the appropriate level of quality and safety and in compliance with construction Code requirements. Therefore, the licensee concluded that the missing Code documentation does not infer a physical lack of quality in the field.

3.7.5 Evaluation and Discussion

In its letter dated April 30, 1999 (ref. 4), the licensee identified documentation deficiencies with 40 pipe welds and 12 hanger-to-pipe welds. Specifically, deficiencies were identified with 22 FPCCS welds that are accessible from the pipes' exterior, 3 CCWS welds that are accessible from the pipes' exterior, and another 15 FPCCS welds that are embedded in concrete and are accessible only from the pipes' interior. The 12 hanger-to-pipe welds are on the exterior surface of the FPCCS piping. Since then, the licensee has replaced three of the accessible FPCCS welds with fully documented Code welds (refs. 4, 8). The licensee provided the staff with a list of the SFPs C & D FPCCS and CCWS welds that had missing documentation and their proposed alternative. The submittals also identified the specific Code requirements that were not satisfied as a result of the missing documentation. In the following paragraphs, the staff evaluated the adequacy of the proposed alternative for the field welds against the objectives of the specific Code requirements associated with the missing documentation.

ND-2150 and ND-4122 require identification and control of pressure-retaining material. These requirements ensure that the appropriate materials are being used for the construction of the FPCCS and CCWS weld and pipe materials. The control of material is part of the licensee's Quality Assurance (QA) program. The licensee stated that a Code-compliant QA program was in existence during the fabrication of the welds with missing documentation. They used one construction program for the entire SFP facility of which SFPs A & B were completed. SFPs A & B are Code-compliant and the Code-required documentation exists. The same work procedures and work control procedures were used for the construction of all four SFPs (A, B, C, & D). After cancellation of Units 2, 3, and 4, the licensee inadvertently discarded field weld documentation for these 49 SFPs C & D field welds. In a letter dated April 30, 1999, the licensee provided specific procedural and program controls that traced the responsibility for verifying the completion of Code requirements to specific personnel. The procedural and

program controls established QA oversight of each step in the construction process that required documentation.

The staff performed an on-site inspection that reviewed past QA procedures, the past construction program, current QA procedures, and current construction program (ref. 17). The purpose of the review of past QA procedures and the construction program was to establish that the documents that are now missing had been completed at the time of construction. The purpose of the review of current QA procedures and construction program was to verify Code compliance of documents recreated by means of re-inspections and the program being used to complete the construction of the SFPs C & D FPCCS and CCWS piping. The review encompassed the records used by the licensee as the bases for the alternative plan and included interviews with past and current Authorized Nuclear Inspectors (ANIs). (The ANI is an independent third party with the responsibility for verifying that construction is being performed according to Code requirements.) The review provided objective evidence of the acceptability of the past and current construction documents and past and current construction programs. The objective evidence consisted of construction and QA records signed off by the ANIs and re-inspections performed as part the alternative plan. Based on the above review of documentation and the on-site inspection, the staff concluded that the objectives of ND-2150 and ND-4122 were satisfied.

ND-4323 requires that only qualified procedures and qualified welders can be used for welding. The licensee stated that they used an ongoing construction program for the entire SFP facility. The same work procedures, work control procedures, welder pool, and welding procedures were used for construction of the four SFPs. During the on-site inspection, the staff reviewed welding records for embedded welds from SFPs A & B that were similar to welds in SFPs C & D. The review included WDRs, welder qualification records, weld quality control inspector records, NDE examiner qualification records, welding procedure specifications (WPS), and procedure qualification records (PQRs) for stainless steel welds. These SFPs A & B construction records were retrievable, legible, and complete. For SFPs C & D, the staff reviewed visual and liquid penetrant re-inspection records and the welder symbols retrieved from accessible welds. The records from the re-inspections and the selected original construction welder qualification records were retrievable and found to be in order. The staff's review provided objective evidence that a detailed quality program was in place and was followed during construction. Based on the data reviewed during the inspection and the existence of an acceptable QA program at the time of construction, the staff concluded that SFPs C & D welds were made by qualified personnel using qualified procedures in accordance with the objectives of ND-4323.

ND-4232.2(b) requires chemical analyses of filler metals or weld deposits be known. In a letter dated April 30, 1999 (ref. 4), the licensee provided information showing chemical checks for 21 accessible FPCCS welds (two of these welds were later replaced with Code welds) and 12 hanger-to-pipe FPCCS welds. The chemical checks were performed on the surface of the welds using an X-ray fluorescence analyzer. During the on-site inspection, the staff observed a performance demonstration of the X-ray fluorescence analyzer's ability to discriminate between different types of stainless steel and chemical extremes within a stainless steel type. The licensee's chemical checks verified that the welds were made from stainless steel filler metal. The licensee corroborated the chemical checks with chemical analyses performed on samples removed from three of the SFPs C & D FPCCS welds. During the inspection (ref. 17), the staff

compared the chromium and nickel from the chemical analyses with filler metals used for similar welds in SFPs A & B, and determined that they were similar and within the ranges of Type 308 stainless steel. The chemical analyses provided objective verification that stainless steel filler metal was used for welding FPCCS piping. Based on the chemical checks, chemical analyses, comparison with SFPs A & B, and an acceptable QA program during construction, the staff concluded that the alternative for the FPCCS welds provided an acceptable level of quality and safety with respect to the chemical composition requirements of ND-2432.2(b).

In a letter dated October 29, 1999 (ref. 9), the licensee provided chemical analyses that were run on samples from the three SFPs C & D CCWS welds. During the on-site inspection, the licensee stated that they used SFA-5.1, E7018 filler metal for welding SFPs C & D CCWS piping. The staff compared the chemical analyses with the filler metal chemistry for SFA-5.1, E7018 and determined that they were similar. Based on the chemical analyses and the existence of a QA program during construction, the staff concluded that the alternative plan for the CCWS welds provided an acceptable level of quality and safety with respect to the chemical composition requirements of ND-2432.2(b).

ND-2433.2 requires that the filler metal for stainless steel have a ferrite number (FN) greater than 5. FN is an indication of resistance to cracking in the weld. In a letter dated October 29, 1999 (ref. 9), CP&L stated that they performed FN tests on 18 accessible FPCCS welds. The test results ranged from 4 to 9 (rounded off). This variation is in agreement with variations listed in paragraph A6.2 to SFA-5.4 of the 1995 Edition of Code which states that a specimen averaging 5.0 percent ferrite (based on data collected from participating laboratories) ranged in FN measurements from 3.5 to 8.0 percent. Paragraph A6.3 to SFA-5.4 of the 1995 Edition of Code states that ferrite variations from weld to weld must be expected due to slight changes in welding and measuring variables. Based on the information from SFA-5.4 and the above test results, the staff considers the FN measurements reasonable, and the measurements support the preceding conclusion that an acceptable austenitic stainless steel filler metal was used for FPCCS welds.

ND-4322.1 requires that welder identification symbol be affixed near their welds. The welder symbol is used for tracing generic implications associated with defective welds. The generic implications are normally detected by nondestructive examinations during construction. The licensee demonstrated that welder symbols were affixed near the welds during construction by retrieving them from the accessible welds for inclusion on new WDRs. For the embedded welds, the welder symbols are not retrievable. As discussed above, the licensee constructed SFPs C & D under a QA program that was in compliance with the Code. Based on the licensee's retrieval of the welder symbols from the accessible welds and their QA program requirements, the staff concluded that welder symbols were properly used. Because the welder identification symbol is a construction QA tool, the staff concludes the objectives of ND-4322.1 were satisfied.

ND-4231 specifies maximum alignment tolerances for pipe welds and the disposition of tack welds in the set-up of the weld joint. As discussed above, the licensee followed a QA program during construction. The QA program includes procedure CQC-19, "Welding Control," Revision 0, with Exhibit 1, "Weld Inspection Checklist." This checklist requires the welding QA/QC inspector to inspect the fit-up, alignment, and the removal of tack welds. To demonstrate that the alignment was in compliance with Code, the licensee performed a visual examination from

the outside surface of all accessible welds and from the inside surface of the embedded welds with a camera. From these examinations, the licensee determined that the fit-up and alignment were acceptable. During the on-site inspection, the staff visually examined the outside surfaces of welds 2-CC-3-FW-207, 2-CC-3-FW-208, and 2-CC-3-FW-209, and determined that the fit-up and alignment looked satisfactory. The staff also looked at fit-up and alignment of the embedded welds recorded on the videotapes and determined that they looked satisfactory.

The staff did not detect evidence of the existence of any tack welds. Therefore, the tack welds were either consumed by the welding process or integrated into the weld. Based on the above discussion, the staff concluded that alignment, fit-up, and tack welds followed Code requirements. The missing documentation for alignment, fit-up, and tack welds have not affected weld quality. Therefore, the staff concludes the welds satisfy the objectives of ND-4231.

ND-4440 requires NDE of the welded surface. The NDE is normally performed on the exterior surface of the welds. As part of the alternative plan, the licensee proposed re-inspecting the exterior surfaces of the accessible welds and the interior surfaces of the embedded welds. A review by the staff of the re-inspection data generated by the licensee on the accessible welds showed that the welds were free of reportable flaws. In the absence of reportable flaws, the surfaces of the accessible welds were demonstrated to satisfy the NDE requirements of ND-4440.

As part of the alternative plan, the licensee examined all 15 embedded-field welds from inside the piping. The examination was performed using an enhanced remote visual technique. In a letter dated October 29, 1999 (ref. 9), the licensee described the visual examination as capable of assessing construction quality, pipe fit-up, pipe alignment, adequacy of a purge during welding, and fusion of the root pass. (An additional benefit of a visual examination is its ability to assess the entire piping system for corrosion and fouling that may have occurred during lay-up.) The licensee performed the visual examinations by sending a mobile video camera with focusing and magnifying capabilities to each embedded weld. The video camera sent images of the piping as it traveled to the welds and of each weld to a television monitor and video recorder. A qualified level II visual examiner viewed the images and recorded any observations. The observations and videotapes were viewed by the level III visual examiner, the ANI, a contractor hired by the licensee (ref. 18), and the staff. Based on the reports generated by the individuals who viewed the observations and videotapes, the staff determined that there were no safety significant imperfections identified by the remote visual examinations. Certain minor imperfections were identified and, based on a detailed review of each, the staff concluded that they were innocuous. The staff concluded that the construction and the current condition of the embedded welds and piping are in accordance with the objectives of ND-4440 requirements.

ND-4452 and ND-4453 require that defects be removed, repaired, and examined. These paragraphs require that the examinations of surface repairs be performed using magnetic particle testing or liquid penetrant testing, and examinations of welded repairs be conducted using the original weld requirements. The licensee stated in their letter dated April 30, 1999 (ref. 4), that repair WDRs for defects identified during in-process welding may also be missing. The assurance that repair WDRs, if any, were completed is supported by the requirements in the licensee's QA program, which stipulates that completed repair WDRs are verified by the QA

inspectors and the ANI. The surface re-inspections performed by the licensee on all accessible welds and the visual inspection of the embedded welds showed no evidence of repairs. Any welds with undocumented repairs, therefore, were viewed from at least one surface. During its on-site inspection, the staff reviewed the licensee's surface inspection efforts and determined that the surface inspections were representative and in order. Based on the preceding discussion, the staff concluded that weld repairs, if any, were properly made and provided an acceptable level of quality.

From the evaluation and discussions above, the staff believes that the actions taken by the licensee to address the 46 SFPs C & D FPCCS welds with missing documentation, and 3 SFPs C & D CCWS welds with missing documentation provide an acceptable level of quality and safety with respect to the original Code construction requirements.

3.7.6 Summary

Based on the above evaluation, the staff concludes that the proposed alternative for the SFPs C & D FPCCS and CCWS welds with missing documentation provide an acceptable level of quality and safety. Pursuant to 10 CFR 50a(a)(3)(i), the staff authorizes the proposed alternative for the 46 SFPs C & D FPCCS welds with missing documentation and 3 SFPs C & D CCWS welds with missing documentation.

3.8 Quality Assurance

3.8.1 Introduction

The licensee's amendment request (ref. 1) included an alternative to certain ASME B&PV Code requirements. The licensee's Alternative Plan consists of two parts. The first part proposes an alternative to ASME documentation requirements, for which required records are missing. The second part proposes to achieve completion of the piping systems in accordance with an equivalent alternative to the original ASME Code requirements.

The formulation of an alternative to the requirements of the original ASME Code was dictated by the licensee's determination that some ASME Code construction documents were inadvertently destroyed. The alternative consists of inspections performed by a variety of examination methods, selective NDE, selective weld replacement, selective chemical analyses, and establishing with reasonable assurance that the missing documents had existed and had been found acceptable by the responsible parties prior to their destruction. In addition, the licensee proposes to complete construction of the ASME Code portion of the plant associated with SFPs C and D in accordance with the its approved 10 CFR Part 50, Appendix B QA program, as supplemented by requirements developed to conservatively reconcile differences from the ASME Code requirements used during original construction.

3.8.2 Background

Construction of HNP Units 1 and 2 was controlled under a single site-wide ASME QA program (ref. 4, Enclosure 5). All four SFPs and the SFPCCS and CCWS for pools A and B were turned over to the plant following completion. The four pools and cooling water systems are described in the HNP FSAR (ref. 13) and have been incorporated into the HNP Unit 1 operating license.

Construction of the cooling system for pools C and D, however, was discontinued following cancellation of HNP Unit 2. By the time HNP Unit 2 was canceled, the majority of mechanical equipment and piping associated with pools C and D was already installed.

Because construction of the cooling water and cleanup system for pools C and D was not complete, construction records were placed in temporary storage. As stated in the licensee's amendment request (ref.1, Enclosure 8), certain piping installation records were inadvertently destroyed. These records included installation verification data and field records for certain ASME Section III piping welds.

3.8.2.1 Regulatory Requirements

GDC 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

Regulatory Guide (RG) 1.26 (Ref. 20) is the principal document used in staff reviews for identifying, on a functional basis, quality standards for nuclear power plant components containing water, steam, or radioactive material. The HNP SFP piping for which certain required records have been lost are ASME Code, Section III, Class 3 components which are classified as "Quality Group C" in accordance with RG 1.26.

Regulatory requirements with respect to the application of the ASME Code for the design, fabrication, erection, construction, testing, and inspection of components for nuclear power plants are specified by §50.55a, "Codes and Standards." Section 50.55a requires that nuclear power plants meet the requirements of the ASME Code, Section III, Division 1 for Class 1, Class 2, and Class 3 components. Alternatives to these requirements may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. For proposed alternatives, the applicant must demonstrate that:

- The proposed alternatives would provide an acceptable level of quality and safety, or
- Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.8.2.2 ASME Code Requirements Addressed by Alternative Plan

The licensee's submittal (ref. 1) provides a detailed description of the proposed alternatives to demonstrate compliance with ASME Code requirements for the cooling and cleanup system piping in accordance with 10 CFR 50.55a(a)(3)(i). The plan consists of two parts. The first part addresses the missing construction documentation associated with the piping installed during original plant construction and intended for the HNP SFP C and D cooling and cleanup system. The second part addresses completion of HNP SFP C and D under an alternative to an ASME N Certificate program.

Field fabrication and installation of the piping covered by the Alternative Plan was performed in accordance with the ASME Code Section III, 1974 Edition, Winter 1976 Addenda (the ASME

Code Edition and Addenda that were required by NRC regulations at the time of issuance of the HNP construction permit). Completion of the piping systems will also be performed in accordance with this edition of the ASME Code, as implemented by the licensee's Alternative Plan.

The ASME Code documentation requirements that cannot be satisfied as a result of the missing records are identified in Enclosure 7 of the licensee's April 30, 1999 RAI response, "Matrix of Code Requirements versus Missing Field Records" (ref. 4, Enclosure 7). The matrix identifies the specific section of the ASME Code, the deficiency, and method of reconciliation.

3.8.2.2.1 Alternative Plan for Missing Construction Records (Pedigree Piping Plan)

The plan addresses the missing construction documentation associated with the portion of the piping completed during original construction. To define the scope of the Alternative Plan, the boundaries of the piping system were determined through documented detailed field walkdowns, during which identification markings such as spoolpiece numbers, welder identification numbers and heat numbers were recorded. As-built isometric drawings were developed from the data collected from the walkdowns. Then, construction-era documents substantiating the quality of the piping were compiled. Where documentation was no longer available, reexamination and testing was performed. Finally, the compilation of original construction records and those generated by reexamination/ testing were reviewed against documentation required by the ASME construction Code. Where ASME Code-required documentation was missing, the body of evidence (e.g., alternate tests or inspections, construction procedures) was evaluated to determine if comparable quality and safety exists. Notably, the missing documentation includes piping isometric packages for approximately 40 of the nearly 200 large bore piping welds in the completed ASME Section III portions of the HNP FPCCS and CCWS for SFP C and D. The alternative plan addresses these 40 large bore field welds and 12 pipe hanger attachment welds.

3.8.2.2.2 Alternative Plan for Continuance of Design and Construction

The original construction of the HNP Nuclear Plant was subject to the full requirements of Section III of the ASME B&PV Code under the authorization of a single N Certificate program maintained by the licensee. The site ASME Section III QA program was discontinued shortly after completion and turnover of HNP Unit 1, and a corporate QA program meeting 10 CFR 50 Appendix B requirements was implemented as required to address plant operation, including implementation of ASME Section XI requirements regarding inspection, repair and replacement activities. Thus, the original construction program no longer exists and it is not possible to complete construction of the C and D FPCCS as a continuance of this program.

The licensee proposes to complete the design of this portion of the plant to meet applicable ASME Section III requirements, but to complete construction under the corporate QA program. Conflicts between the ASME QA program, used during original construction, and corporate QA program requirements will be conservatively dispositioned. The licensee has developed a set of supplemental QA requirements to augment the corporate QA program for completion of Code portions of the plant associated with SFPs C and D. These supplemental requirements were developed by review of the ASME Section III QA program used during original

construction and are intended to conservatively reconcile differences between program requirements.

3.8.3 Evaluation

3.8.3.1 Alternative Plan for Missing Construction Records

This part of the Alternative Plan proposes to demonstrate that the originally installed equipment is acceptable for use. By walkdowns, compilation of original construction records, and by reexamination and testing, the licensee has determined that, except for certain field weld installation records, the piping systems are compliant with applicable ASME Code requirements. Because these records are not available, the cooling and cleanup system for HNP SFPs C and D cannot be demonstrated to have been completed in accordance with the applicable requirements of Section 50.55a.

The welds for which required documentation is missing are identified in the weld matrix submitted as Enclosure 3 to Reference 4. The data was recorded on WDRs, which contained information pertaining to weld attributes, including identification of the items being welded, specification of the weld procedure specification, welder identification, filler metal material identification, nondestructive examination requirements, and signatures attesting to satisfactory completion of activities associated with the welds. These signatures include that of the ANI, an independent third party representing the nuclear insurer, who verifies that construction activities are performed in compliance with ASME Code requirements.

The weld matrix identifies the 40 large bore pipe welds and 12 hanger attachment welds for which records are missing. Of the 40 piping welds, 37 are FPCCS welds (15 of which are embedded in concrete) and three are CCWS welds. All 52 welds were either reinspected or replaced (the 15 welds located on piping embedded in concrete were examined by remote camera). The staff evaluation and acceptance of weld reinspection and testing is addressed in Section 3.7 of this Safety Evaluation. In addition to reinspection and testing, the licensee compiled a substantial body of evidence to substantiate the acceptable quality of these welds. The staff reviewed much of this evidence (Ref. 1, 4, 9) as part of its evaluation, in addition to other plant records made available to NRC inspectors (Ref. 17, 19). The following summary is based on review of this material.

The NRC evaluation of the designs of HNP Unit 1 and Unit 2, including the four SFPs, is reported in NUREG-1038 (ref. 12). The four SFPs are located in the same building, and work was performed simultaneously on all four pools by a common pool of craft, quality control, and engineering resources. Only the cooling systems for pools C and D had not been turned over. The four pools were turned over to the operating organization after completion of construction, and are part of the HNP Unit 1 operating license. The pools and the cooling and cleanup system for SFP A and B are described in Chapter 9 of the FSAR for HNP Unit 1 (ref. 13). Pools A and B have been in operation since plant startup in 1987.

Construction of HNP Units 1 and 2 was controlled under a single site-wide QA program. The ASME QA Program Manual (ref. 4, enclosure 5), effective during the construction period, was reviewed as part of this evaluation. The site-wide QA program and implementing quality procedures effective during the period when major welding activities were ongoing were also

reviewed as part of a special inspection conducted to evaluate the Alternative Plan (Ref. 17). Based on this review, the staff concluded that the QA program provided adequate control of the welding process and that holdpoints were adequate to ensure the quality of welds before proceeding to a subsequent activity. These holdpoints included a detailed review of weld documentation to assure the welds had been completed in accordance with technical, Code, and regulatory requirements. For welds to be embedded in concrete, an additional holdpoint was provided to ensure that welds met all requirements prior to placement of concrete.

In addition to effective control of the welding process by the QA program, construction-era documents provide additional assurance that the subject welds were of acceptable quality. These records include hydrostatic test (hydrotest) records, concrete placement reports, and reports generated by QA personnel.

Hydrostatic tests were verified by QA personnel who verified that all tests, inspections, and documentation required by the Code were complete. The tests were also witnessed by an ANI. In addition to verifying that welds within the scope of a hydrotest met acceptance criteria, the hydrostatic test records also provide evidence that the welds were completed, inspected, and documented in accordance with the licensee's QA program. For the 15 embedded welds that cannot be reinspected directly, the staff reviewed hydrostatic test records for 13 of these welds. For the remaining two welds, the staff reviewed to its satisfaction corrective action documents identifying that these hydrostatic tests had been completed (ref. 17). These records provide evidence that the missing WDRs had been reviewed prior to performance of the hydrostatic tests.

Concrete placement records provide further evidence that the embedded welds were completed in compliance with Code requirements. Since embedding piping in concrete represented a point at which piping was no longer accessible for inspection or rework, procedural controls were established to ensure that all required work activities had been completed and that documentation was in order prior to authorizing concrete placement. For the 15 embedded welds, the staff reviewed and verified that the associated concrete placement reports were properly completed and signed.

Additional supporting documentation related to weld inspection activities is identified in the weld matrix. These documents were reviewed and found to support the conclusion that inspection of activities related to the subject welds was adequate (ref. 17).

In addition to the documents identified in the weld matrix, a sample of quality-related reports were reviewed to assess the effectiveness of QA program implementation. Based on review of these documents, it was concluded that inspection personnel actively monitored welding activities and processes for compliance with ASME and QA program requirements. Deficiencies were accurately reported, appropriately resolved, and corrective actions promptly taken. All corrective action documents reviewed were in compliance with the licensee's QA program and NRC requirements.

Finally, NRC inspection reports during the construction period from 1978 through 1983 were reviewed. This was the period during which the SFPs were under construction. The inspection reports document more than 50 separate inspections for this period for items related to the welding program and/or piping installation. Although several violations dealt with the general

programs, with comments on how these differences are reconciled by the supplemental QA requirements (ref. 4, Enclosures 5, 14, 16, 17). Based on review of these documents conducted as part of this evaluation, the staff has determined that the licensee's alternate approach to §50.55a applicable requirements provides an acceptable level of quality and safety.

3.8.4 Summary

Based on the above evaluation, the staff concludes that, pursuant to the provisions of 10 CFR 50a(a)(3)(i), the alternatives proposed by the licensee for continuance and completion of the piping systems associated with HNP SFPs C and D are acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards consideration if the operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration in its December 23, 1998, amendment request (ref. 1). The staff reviewed the licensee's analysis and, based on its review, it appeared that the three standards of 10 CFR 50.92(c) were satisfied. Therefore, the NRC staff proposed to determine that the amendment request involves no significant hazards consideration, and published its proposed determination in the *Federal Register* for public comment on January 13, 1999 (64 FR 2237).

The staff has completed its evaluation of the licensee's proposed amendment as discussed in Section 3.0 above. Based on its evaluation, the staff has determined that the proposed amendment does not significantly increase the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following staff evaluation in relation to the three standards of 10 CFR 50.92 supports the staff's final no significant hazards consideration determination.

First Standard:

"Involve a significant increase in the probability or consequences of an accident previously evaluated."

The following postulated accidents and events involving spent fuel storage were identified and evaluated by the licensee. The staff likewise evaluated the same accidents and events.

1. a spent fuel assembly drop in a SFP
2. loss of SFP cooling flow
3. a seismic event
4. misloaded fuel assembly

The probability that any of the accidents in the above list can occur is not significantly increased by the proposed changes. The probabilities of a seismic event or loss of SFP cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by procedures and equipment used for handling the fuel. Fuel handling is not a random event; it is strictly controlled using approved procedures, trained personnel, and specialized equipment. Using these methods, the probability of a fuel handling accident (fuel assembly drop or misloading) is minimized, and increasing the number of times it is done will not cause a significant increase in the probability of an accident.

In its submittal, the licensee re-evaluated the consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the SFP. The licensee found that the structural damage to the FHB, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed activity from those considered previously, the structural damage to these items remains unchanged. The staff reviewed the licensee's evaluation in Section 3.3 of this safety evaluation and accepts the licensee's finding. Similarly, the radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. The staff reviewed the licensee's analysis as discussed in Section 3.6 of this safety evaluation. Based on its review, the staff concluded that bounding scenario for the postulated fuel handling accident in the FHB does not change due to the addition of storage racks in SFPs C and D. Therefore, the inputs and assumptions for the dose consequences do not change, and the current fuel handling accident dose assessments in the HNP FSAR (ref. 13) remain bounding. On this basis, the staff concluded that the consequences of this type of previously evaluated accident are not significantly increased by the proposed change.

The staff evaluated the consequences of a loss of SFP cooling event in Section 3.2 of this evaluation. On the basis of their review, the staff determined that sufficient time is available for plant operators to take mitigating actions to restore cooling prior to the pool boiling. In addition, sufficient makeup capability is available should boiling occur. Thus, the consequences of this type of accident are not significantly increased from previously evaluated loss of cooling events.

The staff evaluated the consequences of a design basis seismic event in Section 3.4 of this evaluation. On the basis of their review, the staff concluded that the licensee's structural analysis and design of the spent fuel rack modules is acceptable and that the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14)., and are, therefore, acceptable. Thus, the consequences of a seismic event are not significantly increased from previously evaluated events.

The staff evaluated the consequences of fuel misloading and mislocation events in Section 3.1 of this evaluation. In their evaluation, the staff states that while most abnormal storage conditions will not result in an increase in the k-eff of the racks, its is possible to postulate events which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of soluble boron in the pool water based upon the double contingency principle which requires at least two unlikely, independent, concurrent events to occur before a

nuclear criticality accident is possible. Therefore, since soluble boron is normally present in the SFP water, credit for soluble boron may be assumed in evaluating other accident conditions such as the misloading of fresh fuel. Plant procedure CRC-001 requires that the soluble boron concentration in the pool be maintained between 2000 and 2600 ppm and is confirmed by monthly surveillance measurements. The negative reactivity credited to the boron more than offsets the reactivity addition caused by credible accidents. In fact, Holtec has determined that a soluble boron concentration of only 400 ppm would be sufficient to maintain k-eff less than 0.95 even if a fresh PWR assembly were inadvertently placed in a location restricted to a burned assembly as per TS Figure 5.6.1. Thus, the consequences of fuel misloading and mislocation events are not significantly increased from previously evaluated events.

Second Standard:

"Create the possibility of a new or different kind of accident from any previously analyzed."

As noted in various sections of this safety evaluation, the staff evaluated the proposed changes in accordance with appropriate NRC Regulatory Guides, SRP sections, and industry codes and standards. In addition, the staff has previously prepared several safety evaluations for rerack applications which are similar to this proposal. No unproven techniques or methodologies were used in the analysis and design of the racks to be used in SFPs C and D. No unproven technology will be used in the installation of the racks.

In its analysis, the licensee conservatively considered an accidental drop of a rack module during construction activity in SFPs C and D as an event which might represent a new or different kind of accident. However, this event had been considered previously by the licensee for rack installation in SFP B, which used the same equipment and procedures that will be used for installation activities in SFPs C and D.

The staff evaluated the handling of heavy loads and spent fuel assemblies in Section 3.3 of this evaluation. On the basis of their review, the staff determined that the licensee's use of the 10-ton single-failure-proof FHB auxiliary crane, the 20-ton hoist, the spent fuel rack lifting rig, and administrative controls and procedures that are in accordance with NUREG-0612 and ANSI N14.6, will help to maintain safety during the installation of the new racks. The reliability of the crane coupled with the design, testing and inspection of the crane, the lifting rig and other lifting devices will enable the licensee to handle safely the racks and other heavy loads during the rack installation process. The postulated accident analyses involving a dropped spent fuel storage rack and SFP gate indicated that the SFP liner could be breached. However, during such a breach, the licensee could maintain the pool and its contents within the acceptable consequence limits set forth in NUREG-0612. In addition, the licensee's use of administrative controls and procedures to improve the handling and control of heavy loads, including the racks, enhances the licensee's capability to reduce the potential for load drops. In addition, the staff reviewed the licensee's spent fuel shipping cask handling operations and determined that they will not occur over the SFPs. Therefore, movement of the cask will not impact the new racks nor the stored spent fuel configuration in the pools. Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any previously analyzed.

Third Standard:

"Involve a significant reduction in the margin of safety."

The function of the Spent Fuel Pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with Pools C and D.

In evaluating a potential reduction in the margin of safety, the licensee addressed the safety issues related to the expanded pool storage capacity in the following areas. The staff likewise evaluated the same areas.

1. material, mechanical and structural considerations
2. nuclear criticality considerations
3. thermal-hydraulic and pool cooling considerations

The staff evaluated the material, mechanical and structural considerations of the proposed amendment in Sections 3.4 and 3.5 of this evaluation. Based on its evaluation the staff determined that the materials used in the fabrication of the spent fuel racks manufactured by Holtec are compatible with the SFP environment at HNP and that the type of degradation exhibited by the racks does not affect their neutron-absorbing capability. The staff concluded, therefore, that the materials used in the new spent fuel racks are acceptable. With respect to mechanical and structural considerations, the staff concluded that the licensee's structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads, including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR (ref. 13) and applicable provisions of the SRP (ref. 14), and are, therefore, acceptable. Thus there is no significant reduction in margin of safety related to the material, mechanical and structural considerations.

The staff evaluated the nuclear criticality aspects of the proposed amendment in Section 3.1 of this evaluation. The NRC acceptance criterion for subcriticality is that the effective multiplication factor (k-eff) in the spent fuel pool storage racks when fully flooded by unborated water shall be no greater than 0.95, including uncertainties at a 95 percent probability, 95 percent confidence level (95/95) under all conditions. On the basis of their review, the staff determined that the analysis methods used are acceptable and capable of predicting the reactivity of the HNP storage racks with a high degree of confidence. Therefore, the staff concluded that the criticality aspects of the proposed storage capacity expansion for HNP spent fuel pools C and D are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. Thus there is no significant reduction in margin of safety related to nuclear criticality considerations.

The staff evaluated the thermal-hydraulic and pool cooling aspects of the proposed amendment in Section 3.2 of this evaluation. On the basis or their review, the staff determined that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure. In addition, given the decay

load in SFPs C and D will be limited to 1 Mbtu/hr, the staff concluded that sufficient time is available for plant operators to take mitigating actions prior to pool boiling in the event of a loss of SFP cooling. Thus, there is no significant reduction in margin of safety related to thermal-hydraulic and pool cooling considerations.

On the basis of the above evaluation, the NRC has made a final determination that the proposed amendment does not involve a significant hazards consideration.

5.0 COMMENTS ON PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Several comments were received in response to the staff's January 13, 1999, proposed no significant hazards consideration determination (64 FR 2237). These comments and the staff's response are grouped into subject area categories and addressed below.

5.1 Risk/Accident Concerns

Comment - Several comments questioned how the probability or consequences of fuel handling accidents would not be significantly increased if the amount of fuel being moved and stored would be more than double what is currently licensed.

Response - As described in Section 4 above, the risk of a fuel handling accident is primarily influenced by procedures and equipment used for handling the fuel. Fuel handling is not a random event; it is strictly controlled using approved procedures, trained personnel, and specialized equipment. Using these methods, the probability of a fuel handling accident is minimized, and increasing the number of times it is done will not cause a significant increase in the probability of an accident.

As for the consequences of an accident, the fuel handling accident analysis for HNP was re-evaluated for the proposed changes. The resulting consequences are bounded by the existing analysis since the pertinent fuel parameters remain unchanged.

5.2 Cooling System Capacity

Comment - Several comments questioned the capacity of the existing HNP cooling systems to support the additional heat load of SFPs C and D.

Response - The staff's evaluation of the adequacy of the cooling system is discussed in section 3.2 above. Based on its review the staff concluded that there is sufficient thermal margin in the CCW and ESW systems to maintain the bulk fuel pool coolant temperature in all SFPs within their design limits, assuming an additional decay heat load of 1 Mbtu/hr in SFPs C and D, and assuming a single active failure.

5.3 Piping Concerns

Comment - Several comments questioned the adequacy of CP&L's alternative plan for demonstrating that the CCW and FPCCS piping to support SFPs C and D is capable of performing its design function.

Response - The licensee's alternative plan is discussed in detail in sections 3.7 and 3.8 above. Based on its review, the staff concluded that the proposed alternative for the SFPs C and D FPCCS and CCW system welds with missing documentation provides an acceptable level of quality and safety.

5.4 Transportation Concerns

Comment - A few comments questioned the safety of transporting spent fuel from CP&L's other two reactor sites (Brunswick and Robinson) for storage at HNP.

Response - The Operating License for HNP, which was issued in January 1987, authorized CP&L to receive and store fuel from Brunswick and Robinson at HNP. This amendment does not change that approval. CP&L can continue to ship fuel from its other sites using approved procedures as it has in the past.

5.5 Requests for Public Meeting/Hearing

Comment - Several comments requested that public hearings/meetings be held to discuss CP&L's proposed amendment.

Response - In response to the January 13, 1999, *Federal Register* Notice, BCOC filed a request for a hearing. As discussed previously, an ASLB panel was formed and BCOC was granted its hearing request. As part of the hearing process, a public pre-hearing conference was held in Chapel Hill on May 13, 1999, and Limited Appearance Statements were heard on December 7 and 8, 1999, in both Raleigh and Chapel Hill, North Carolina. The ASLB heard oral arguments on the admitted technical contentions on January 21, 2000, in Rockville, Maryland. The ASLB also heard oral arguments on the environmental contention on December 7, 2000, in Raleigh, North Carolina. The environmental contention was admitted by the ASLB on August 7, 2000, in response to BCOC's January 31, 2000, filing.

In addition, in response to the requests for a meeting, the staff held a public meeting in Raleigh, North Carolina, on February 28, 2000, to discuss its license amendment review process, and the scope and status of its review of CP&L's request. The majority of this meeting was used for public comments and questions. The staff issued a summary of this meeting on April 19, 2000.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the *Federal Register* on December 21, 1999 (64 FR 71514). Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

Amendment Related

1. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Request for License Amendment Spent Fuel Storage," dated December 23, 1998. (Accession number 9812290056)

2. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Spent Fuel Storage Re-Designation of Proprietary Information," dated March 15, 1999. (Accession number 9903220064)

3. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Spent Fuel Storage - Page Additions - Holtec Report," dated April 5, 1999. (Accession number 9904130221)

4. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the Alternative Plan for Spent Fuel Pools Cooling and Cleanup System Piping," dated April 30, 1999. (Accession number 9905050200).

5. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the License Amendment Request to Place HNP Spent Fuel Pools C & D in Service," dated June 14, 1999. (Accession number 9906210117).

6. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding Amendment Request to Increase Fuel Storage Capacity by Placing Spent Fuel Pools C & D in Service," dated July 23, 1999. (Accession number 9907270169).

7. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding Amendment Request to Increase Fuel Storage Capacity," dated September 3, 1999. (Accession number 9909100158).

8. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Supplemental Information Regarding the License Amendment to Place HNP Spent Fuel Pools C & D in Service," dated October 15, 1999. (Accession number 9910270013).

9. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding the Alternative Plan for Spent Fuel Pools C & D Cooling and Cleanup System Piping," dated October 29, 1999. (Accession number ML993160242).

10. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information (RAI) Regarding Rack Installation Spent Fuel Pools C & D," dated April 14, 2000. (Accession number ML003707409).

11. Letter from Carolina Power & Light Company to U. S. Nuclear Regulatory Commission, "Supplemental Changes to License Amendment Request - Spent Fuel Storage," dated July 19, 2000. (Accession number ML003734906).

12. NUREG 1038, "Safety Evaluation Report related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2," dated November 1983. Supplement 1 dated June 1984; Supplement 2 dated June 1985; Supplement 3 dated May 1986; Supplement 4 dated October 1986.

13. Final Safety Analysis Report - Shearon Harris Nuclear Power Plant.

14. NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition."

15. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Rev. 1, December 1975 (for comment); Rev. 2, December 1981 (for comment).

16. "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.

17. NRC inspection Report No.: 50-400/99-12, December 28, 1999. (Accession number ML003673416)

18. G. J. Licina, "Evaluation of Embedded Welds in Spent Fuel Piping at Harris Nuclear Plant," Structural Integrity Associates, Inc., San Jose, CA, Report No. SIR-99-127, Rev. 0 and Rev. 2 dated December 1999. (Accession number ML003712432 (Rev. 0); ML003712470 (Rev. 2).

19. NRC inspection Report No.: 50-400/2000-05, February 16, 2000. (Accession number ML003685113)

20. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants," Revision 3, February 1976.

21. Regulatory Guide 1.33, Revision 2, "Quality Assurance Program Requirements (Operation), February 1978.

22. Regulatory Guide 1.88, Revision 2, "Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records," October 1978.

ASLB Hearing Related

23. Orange County's Request for Hearing and Petition to Intervene, February 12, 1999. (Accession number 9902250036)

24. Orange County's Supplemental Petition to Intervene, April 5, 1999. (Accession number 9904080103)

25. NRC Staff's Response to Orange County's Supplemental Petition to Intervene, May 5, 1999. (Accession number 9905070018)

26. Applicant's Answer to Petitioner Board of Commissioners of Orange County Contentions, May 5, 1999. (Accession number 9905100006)

27. Transcript of Pre-Hearing Conference RE: Carolina Power & Light Company, May 13, 1999. (Accession number 9905200044)

28. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Standing and Contentions), July 12, 1999. (Accession number 9907130054)

29. ASLBP No. 99-762-02-LA; Memorandum and Order (Granting Request to invoke 10 CFR Part 2, Subpart K Procedures and Establishing Schedule), July 29, 1999. (Accession number 9908020101)

30. Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant, January 4, 2000. (Accession number ML003672688)

31. Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely at the Subpart K Oral Argument, January 4, 2000. (Accession number ML003672886)

32. NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Arguments on Technical Contentions 2 and 3, January 4, 2000. (Accession number ML003673204)

33. Transcript of Oral Arguments RE: Carolina Power & Light Company, January 21, 2000. (Accession number ML003679424)

34. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing), May 5, 2000 (LBP-00-12). (Accession number ML003712091)

35. Orange County's Request for Admission of Late-Filed Environmental Contentions, January 31, 1999[2000]. (Accession number ML003680571)

36. Carolina Power & Light Company; Shearon Harris Nuclear Power Plant, Unit 1, Environmental Assessment and Finding of No Significant Impact; Federal Register, December 21, 1999 (64 FR 71514)

37. Applicant's Response to BCOC's Late-Filed Environmental Contentions, March 3, 2000. (Accession number ML003690965)

38. NRC Staff Response to Intervenor's Request for Admission of Late-Filed Environmental Contentions, March 3, 2000. (Accession number ML003690415)

39. Orange County's Reply to Applicant and Staff's Oppositions to Request for Admission of Late-Filed Environmental Contentions, March 13, 2000. (Accession number ML003694928)

40. Orange County's Petition for Review of LBP-00-12, May 22, 2000. (Accession number ML003719114)

41. NRC Staff Response to Orange County's Petition for Review of LBP-00-12, June 6, 2000. (Accession number ML003735783))

42. Applicant's Answer Opposing Commission Review of LBP-00-12, June 6, 2000. (Accession number ML003723761)

43. Commission Memorandum and Order (CLI-00-11) in response to BCOC's May 22, 2000, Petition, June 20, 2000. (Accession number ML003725065)

44. ASLBP No. 99-762-02-LA; Memorandum and Order (Ruling on Late-Filed Environmental Contentions), August 7, 2000 (LBP-00-19). (Accession number ML003738358)

45. Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with respect to the need to prepare an Environmental Impact Statement to Address the Increased Risk of a Spent Fuel Pool Accident, November 20, 2000. (Accession number ML003772525)

46. Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely at the Subpart K Oral Argument Regarding Contention EC-6, November 20, 2000. (Accession number ML003771805)

47. NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon which the Staff Proposes to Rely at Oral Arguments on Environmental Contention 6, November 20, 2000. (Accession number ML003771530)

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Date: December 21, 2000

advance of the holding or completion of any required hearing, where it has determined that no significant hazards considerations are involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards considerations. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendment has been issued and made immediately effective and any hearing will be held after issuance.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (64 FR 71514).

For further details with respect to the action see (1) the application for amendment dated December 23, 1998, as supplemented on March 15, April 5, April 30, June 14, July 23, September 3, October 15, and October 29, 1999, and April 14, and July 19, 2000, (2) Amendment No. 103 to License No. NPF-63, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 21st day of December 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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NAME	RLaufer	Dunnington	SUttal *	RCorreia *	
DATE	11/20/00	11/20/00	8/15/00	11/16/00	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

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Cite as 52 NRC 85 (2000) LBP-00-19

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Frederick J. Shon
Dr. Peter S. Lam

In the Matter of

Docket No. 50-400-LA
(ASLBP No. 99-762-02-LA)

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant)

August 7, 2000

In this 10 C.F.R. Part 2, Subpart K proceeding, ruling on the admissibility of four late-filed contentions challenging the NRC Staff's environmental assessment determination not to prepare a environmental impact statement (EIS) under the National Environmental Policy Act of 1969 (NEPA) regarding Applicant Carolina Power & Light Company's (CP&L) request to increase the spent fuel storage capacity of its Shearon Harris Nuclear Power Plant through a 10 C.F.R. § 50.90 facility operating license amendment, the Licensing Board finds one of the contentions admissible under the 10 C.F.R. § 2.714(a)(1), (b), and (d) standards governing late-filed issues and establishes a schedule for its further litigation.

RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS

Relative to late-filed contentions, it is well established that the burden rests with the petitioner to address affirmatively all five factors and demonstrate that, on balance, they warrant excusing the lateness of the filing. Moreover, even if

a late-filed contention fulfills the section 2.714(a)(1) requirements, it must still satisfy the admissibility standards set forth in section 2.714(b)(2)(i)-(iii), (d)(2), in order to receive merits consideration. *See, e.g., Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-99-43, 50 NRC 306, 312 (1999) (citing cases), *petition for interlocutory review denied,* CLI-00-2, 51 NRC 77 (2000).

RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS (GOOD CAUSE FOR DELAY)

It is, of course, also well established that the first section 2.714(a)(1) factor — whether there is "good cause" for the failure to file on time — is the most important component in the late-filed balancing equation.

RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS

Relative to the section 2.714(a)(1) late-filing test, among the four non-good cause factors three and five — assistance in developing a sound record and broadening the issues and delaying the proceeding — are given more weight in the balancing process. *See Commonwealth Edison Co.* (Braidwood Nuclear Power Station, Units 1 and 2), CLI-86-8, 23 NRC 241, 244-45 (1986).

NEPA: REQUIREMENT FOR IMPACT STATEMENT; REMOTE AND SPECULATIVE EVENT

The standard for requiring that an EIS be prepared is whether the action at issue is a major federal action having a significant impact on the human environment. Further, the agency is not required to address in an EIS consequences of an action that are "remote and speculative."

NEPA: CONSIDERATION OF SEVERE ACCIDENTS; REMOTE AND SPECULATIVE EVENT

Over the past decade the Commission has come to rely on probabilistic analysis ever more heavily in the process of making decisions. Indeed, the entire trend in licensing, enforcement, inspection, and the granting of amendments has swung gradually toward decision-making by probabilistic risk assessment. In the NEPA context, deciding what is "remote and speculative" by examining the probabilities inherent in a proposed accident scenario is thus appropriate.

NEPA: SUFFICIENCY OF CONTENTIONS (SABOTAGE)

Assertions regarding sabotage risk do not provide a litigable basis for a contention asserting that an environmental impact statement should be prepared for a spent fuel pool expansion request. *See Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 701 (1985), *review declined,* CLI-86-5, 23 NRC 125 (1986), *aff'd, Limerick Ecology Action v. NRC,* 869 F.2d 719, 744 (3d Cir. 1989).

RULES OF PRACTICE: DISCOVERY (AGAINST ADVISORY COMMITTEE ON REACTOR SAFEGUARDS)

Any attempt to obtain discovery materials or testimony from Advisory Committee on Reactor Safeguards (ACRS) members, staff, or consultants is subject to the exceptional circumstances showing of 10 C.F.R. § 2.720(h). *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-519, 9 NRC 42, 43 n.2 (1979).

MEMORANDUM AND ORDER
(Ruling on Late-Filed Environmental Contentions)

Pending before the Licensing Board is the motion of Intervenor Board of Commissioners of Orange County, North Carolina (BCOC), seeking admission of four late-filed contentions. Each of these issue statements concerns the purported need for the NRC Staff to prepare an environmental impact statement (EIS) regarding the pending request of Applicant Carolina Power & Light Company (CP&L) for an amendment to its operating license for its Shearon Harris Nuclear Power Plant (Harris) to permit the addition of rack modules to spent fuel pools (SFPs) C and D and to place those pools in service. Although both CP&L and the Staff declare that a balancing of the five late-filing elements of 10 C.F.R. § 2.714(a) weighs in favor of admitting the contentions, they nonetheless assert that the contentions should be rejected as lacking adequate basis and specificity as required by section 2.714(b), (d).

For the reasons set forth below, we find that (1) the section 2.714(a) balancing process supports admission of the contentions notwithstanding their "lateness"; and (2) one of the environmental contentions, which we redesignate as Environmental Contention (EC)-6, should be admitted, subject to the limitations described herein. Additionally, we establish a schedule for the further litigation of contention EC-6.

I. BACKGROUND

The question of the admission for litigation of the general subject matter of the four late-filed contentions now before the Board first arose in the context of BCOC's initial, timely filed contentions. In its April 5, 1999 supplement to its February 1999 hearing petition, BCOC proffered five issue statements, which were designated EC-1 through EC-5, challenging CP&L and Staff compliance with the requirements of the National Environmental Policy Act of 1969 (NEPA) relative to the Applicant's SFP expansion amendment. Among other things, those contentions asserted that the proposed license amendment was not exempt from NEPA's requirements under 10 C.F.R. § 51.22; that an EIS was required that addressed amendment effects on Harris accident probability and consequences and alternative costs and benefits, including severe accident mitigation design alternatives (SAMDAs) and dry cask storage; that the EIS needed to address storage of spent fuel from CP&L's Brunswick and Robinson plants; that an environmental assessment must be conducted; and that a discretionary EIS is required under 10 C.F.R. §§ 51.20(b)(14), 51.22(b). As we described in our July 1999 memorandum and order ruling on the admissibility of those five contentions, as a result of a superseding Staff determination to prepare an environmental assessment (EA) relating to the proposed CP&L license amendment, we concluded BCOC's concerns were premature and dismissed those contentions, albeit without prejudice to their being raised at a later juncture, as appropriate. *See* LBP-99-25, 50 NRC 25, 38-39 (1999).

In that same issuance, we admitted two of BCOC's technical contentions that thereafter were subject to litigation in accordance with the provisions of 10 C.F.R. Part 2, Subpart K. While the parties were preparing for 10 C.F.R. § 2.1113 oral presentations to the Board on the issue of whether there were disputed material facts that warranted further exploration in an evidentiary hearing relative to the admitted BCOC technical contentions, the Staff provided the Board and the other parties with a Board Notification indicating that on December 15, 1999, it had issued an EA regarding the CP&L amendment request. *See* Letter from Richard J. Laufer, Project Manager, NRC Office of Nuclear Reactor Regulation to Licensing Board and Parties (Jan. 10, 2000). In its EA, which was published in the Federal Register on December 21, 1999, the Staff concluded that an EIS was unnecessary relative to the CP&L spent fuel pool expansion request because it did not involve a proposed action that would have a significant effect on the quality of the human environment. *See* 64 Fed. Reg. 71,514, 71,516 (1999).

Relative to this EA, on January 31, 2000, BCOC filed the request for admission of four late-filed NEPA-related contentions that is now pending with the Board. In these contentions, which are numbered EC-1 through EC-4, BCOC challenges the Staff's EA, asserting that (1) an EIS must be prepared because the proposed Harris SFP expansion would create accident risks substantially in excess of those the Staff identified in the EA or previously evaluated in the Harris operating license EIS that would significantly affect the quality of the human environment; (2) the EIS that must be prepared must evaluate the significant cumulative environmental risk posed by the operation of pools A, B, C, and D that was not acknowledged in the EA; (3) the EIS that must be prepared must include within its scope an analysis of the impacts of storage of spent fuel from the Brunswick and Robinson nuclear power plants; and (4) a discretionary EIS is needed. BCOC further asserts that a balancing of the five late-filing elements of 10 C.F.R. § 2.714(a)(1) supports a finding that the timing of its filing should not be a bar to their admission. Additionally, BCOC provides information regarding the grounds for each contention that it declares is sufficient to provide the requisite specificity and basis in accordance with the substantive contention admission standards in section 2.714(b), (d). *See* [BCOC] Request for Admission of Late-Filed Environmental Contentions (Jan. 31, 2000) at 23-27 [hereinafter BCOC Contentions Request].

On March 3, 2000, CP&L and the Staff filed responses to the BCOC late-filed request. Both assert that section 2.714(a) late-filing factors three and five — developing a sound record and broadening or delaying the proceeding — do not support late-filed admission. In particular, both suggest relative to factor three that BCOC supporting affiant Dr. Gordon Thompson lacks the requisite education, qualifications, and experience to assist the Board in developing a sound record. Neither, however, contests that BCOC has established that the paramount "good cause" factor, along with factors two and four — availability of other means or parties to protect BCOC's interests — all weigh in favor of admitting the contentions, thereby tipping the overall balance in favor of a finding that late-filing does not bar admission of the contentions. *See* [CP&L] Response to BCOC's Late-Filed Environmental Contentions (Mar. 3, 2000) at 1-2 [hereinafter CP&L Contentions Response]; NRC Staff Response to [BCOC] Request for Admission of Late-Filed Environmental Contentions (Mar. 3, 2000) at 1-4 [hereinafter Staff Contentions Response].

What CP&L and the Staff do dispute is BCOC's claim that the contentions fulfill the pleading requirements of section 2.714, asserting for various reasons that each of the contentions lacks the requisite specificity and basis. *See* CP&L Contention Response at 7-29; Staff Contention Response at 7-29. In a March 13, 2000 reply to the CP&L and Staff responses, BCOC challenges their claims regarding the adequacy of Dr. Thompson's qualifications relative to late-filing factor three as well as their assertions concerning the adequacy of the four contentions. *See* [BCOC] Reply to [CP&L's] and Staff's Oppositions to Request for Admission of Late-Filed Environmental Contentions (Mar. 13, 2000) at 1-22 [hereinafter BCOC Contentions Reply].

Subsequently, it came to the Board's attention that there was outstanding on the public record a recent draft Staff technical study concerning spent fuel pool accident risks, *see* 65 Fed. Reg. 8752 (2000) (soliciting public comment on draft

report), which was one of the matters that was of concern to BCOC in the context of its contention denominated as EC-1, Environmental Impact Statement Required. Although recognizing that this Staff report dealt with spent fuel pool accident risks associated with facility decommissioning activities, the Board provided the parties with an opportunity to provide their views, and respond to the views of the other parties, on the relevance, if any, of this study to the issues before the Board. *See* Memorandum and Order (Requesting Additional Information) (Mar. 21, 2000) at 1-2 (unpublished). Thereafter, all three of the parties filed comments regarding the draft Staff report. BCOC asserted that although the study's limited scope — i.e., decommissioning — restricted its relevance, the Staff's technical analysis still was pertinent in that it (1) further illustrates how the Staff has underestimated the risks of SFP accidents because that study does not include an assessment of the phenomena associated with partial exposure of fuel assemblies, a subject that is at the center of Dr. Thompson's concerns about the SFP accident risks; (2) fails to consider the effect of fuel age on potential for propagation of exothermic reactions; (3) does not discuss criticality accident risk from the placement of low-burnup fuel in a pool in which there is reliance on burnup credit to prevent criticality; and (4) lacks sufficient information regarding zirconium fire propagation. *See* [BCOC] Response to Board's Information Request (Mar. 29, 2000) at 2-10; *see also* [BCOC] Reply to [CP&L's] and Staff's Responses to Board's Information Request (Apr. 5, 2000) at 2-7. Both CP&L and the Staff, on the other hand, found the draft report basically irrelevant to the admission of the contention because it concerns a decommissioned reactor rather than an operating reactor like Harris, although each found points in the draft report, such as the availability and timing of pool water makeup, that supported its position that BCOC contention EC-1 was not admissible. *See* [CP&L] Response to Board's Request Regarding Relevance of Staff's Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants (Mar. 29, 2000) at 2-6; NRC Staff Response to the Atomic Safety and Licensing Board's Request for Additional Information (Mar. 29, 2000) at 2-5; *see also* CP&L Reply to Parties' Responses Regarding Relevance of Staff's Draft Decommissioning Study (Apr. 5, 2000) at 2-3; NRC Staff's Reply to [BCOC] Response to the Board's Request for Additional Information (Apr. 5, 2000) at 2-5.

Thereafter, by order dated May 5, 2000, the Board again requested information from the parties in connection with the draft Staff report, prompted by an April 13, 2000 public record letter from Advisory Committee on Reactor Safeguards (ACRS) Chairman Dana A. Powers to NRC Chairman Richard A. Meserve providing ACRS views on the draft Staff report, including concerns about the potential for exothermic reactions in the event a pool is drained and the resulting release of rutherium, as a source term element. *See* Licensing Board Memorandum and Order (Requesting Additional Information) (May 5, 2000) at 1-2 (unpublished). In its May 15, 2000 response, BCOC found this letter

reinforced its contention EC-1 claim that spent fuel pool accident risks are greater than the Staff assumes because the Staff does not understand the potential for SFP exothermic reactions. *See* [BCOC] Response to May 5, 2000, Memorandum and Order (Requesting Additional Information) (May 15, 2000) at 1-4. In their May 15 responses, CP&L and the Staff maintained that, like the Staff draft report, the ACRS letter is irrelevant because it deals with a decommissioned facility, not an operating reactor like Harris. *See* [CP&L] Response to Board's Request Regarding Relevance of ACRS Letter Addressing NRC Staff Draft Decommissioning Study (May 15, 2000) at 1-3; NRC Staff Response to the Atomic Safety and Licensing Board's Second Request for Additional Information (May 15, 2000) at 2-3; *see also* [CP&L] Reply to Parties' Responses Regarding Relevance of ACRS Letter Addressing NRC Staff Draft Decommissioning Report (May 22, 2000) at 2-5; NRC Staff Reply to [BCOC] Response to May 5, 2000, Memorandum and Order (Requesting Additional Information) (May 22, 2000) at 1-2.

Finally, in response to a July 12, 2000 BCOC motion, on July 13, 2000, the Board granted leave for the parties to comment on a June 20, 2000 letter from ACRS Chairman Powers to NRC Chairman Meserve concerning the proposed resolution of outstanding Generic Safety Issue (GSI)-173A, regarding an action plan for resolving issues relating to operating reactor SFPs. *See* Licensing Board Memorandum and Order (Granting Motion for Leave to Comment) (July 13, 2000) at 1-2 (unpublished). BCOC took the position that, as with the ACRS comments on the Staff decommissioning study, this letter was relevant to its accident risk contention, particularly as it concerns SFP radiological inventories and release characteristics. *See* [BCOC] Comments on Relevance of June 20, 2000, ACRS Letter with Respect to Pending Environmental Contentions (July 20, 2000) at 3-4. In their comments on the ACRS letter, both CP&L and the Staff asserted that this ACRS letter had no relevance to the BCOC contentions because, as with the previous ACRS letter, it does not concern that specific beyond-design-basis reactor accident scenario that is the underpinning for the BCOC accident risk contention. *See* [CP&L] Comments on Relevance of June ACRS Letter to Pending Environmental Contentions (July 20, 2000) at 3-8; NRC Staff Comments on [ACRS] Letter of June 20, 2000 (July 20, 2000) at 2-3; *see also* [CP&L] Reply to Parties' Comments on Relevance of June ACRS Letter to Pending Environmental Contentions (July 27, 2000) at 2-4.

II. ANALYSIS

All the parties recognize that the five late-filing factors set forth in 10 C.F.R. § 2.714(a)(1) are applicable to BCOC's four pending environmental contentions. And, relative to such late-filed contentions, it is well established that the burden rests with the petitioner, here BCOC, to address affirmatively all five factors

and demonstrate that, on balance, they warrant excusing the lateness of the filing. Moreover, even if a late-filed contention fulfills the section 2.714(a)(1) requirements, it must still satisfy the admissibility standards set forth in section 2.714(b)(2)(i)-(iii), (d)(2), in order to receive merits consideration. *See, e.g., Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), LBP-99-43, 50 NRC 306, 312 (1999) (citing cases), *petition for interlocutory review denied,* CLI-00-2, 51 NRC 77 (2000).

A. Application of 10 C.F.R. § 2.714(a)(1) Late-Filing Criteria

It is, of course, also well established that the first factor — whether there is "good cause" for the failure to file on time — is the most important component in the late-filed balancing equation. The BCOC environmental contentions now at issue were not filed until some 9 months after contentions were due in this proceeding. Nonetheless, section 2.714(b)(2)(iii) recognizes that a petitioner can file amended or new contentions "if there are data or conclusions in the NRC draft or final environmental impact statement, environmental assessment, or any supplements relating thereto, that differ significantly from the data or conclusions in the applicant's [environmental report]." Here, the crux of BCOC's concerns, as expressed in its January 2000 contentions, is that the Staff erred in its December 1999 EA in concluding that no EIS is needed. As both CP&L and the Staff acknowledge, there is good cause for such a "late-filed" challenge, assuming the contentions involved are filed within a reasonable time after BCOC became, or should have become, aware of the Staff EA.

In this instance, BCOC's late-filed contentions pleading was submitted some 45 days after the EA was first provided to BCOC counsel by fax from the Staff. BCOC declares that this period for filing was reasonable given that BCOC counsel (1) until January 4, 2000, was involved in preparing its 10 C.F.R. § 2.1113 written presentation regarding the two admitted technical contentions; (2) between January 8 and January 17, was on previously scheduled, 10-day overseas nonvacation trip; (3) between January 17 and January 21, was involved in preparing for and participating in the oral argument regarding that filing, which was held during an all-day session on January 21, 2000; and (4) between January 24 and January 31, was working on two other cases, and was out of her Washington, D.C. office on 1 day and was unable to reach her client on 2 days because of inclement weather. *See* BCOC Contentions Request at 23-25. Neither CP&L nor the Staff disputes that, under the circumstances, the "good cause" element of the section 2.714(a)(1) test has been fulfilled such that this factor favors admitting the contentions. We agree, and thus place this central factor on the "acceptance" side of the balance.

Relative to the other four factors, we also agree with the parties that factors two and four — availability of other means to protect petitioner's interests and extent of representation of petitioner's interests by other parties — weigh in BCOC's favor. As to factors three and five, which among the four non-good cause elements are given more weight in the balancing process, *see Commonwealth Edison Co.* (Braidwood Nuclear Power Station, Units 1 and 2), CLI-86-8, 23 NRC 241, 244-45 (1986), both are problematic in terms of their impact on the balance. Given our May 2000 ruling in favor of CP&L on the two technical issues we admitted for merits consideration, *see* LBP-00-12, 51 NRC 247, *petition for review denied as interlocutory,* CLI-00-11, 51 NRC 297 (2000), the admission of any of these environmental contentions undoubtedly will broaden the issues and delay the proceeding. Moreover, relative to element three — assistance in developing a sound record — our observation in our May 2000 decision that Dr. Thompson's expertise on reactor technical issues appeared to be "largely policy-oriented rather than operational" does not render this a compelling element on BCOC's side of the balance. Nonetheless, in the circumstances here, these two negative elements are not sufficient to overcome the combined weight of factors one, two, and four as supporting a finding that the late-filing of these contentions does not bar their admission.

B. Application of 10 C.F.R. § 2.714(b), (d) Admissibility Criteria

In determining whether the four BCOC environmental contentions are admissible in accordance with the standards set forth in section 2.714(b) and (d), we note initially that we previously dismissed contentions denominated as EC-1 through EC-5 in our July 1999 ruling on BCOC's standing and the admissibility of its timely filed contentions. Three of the four BCOC late-filed contentions essentially track these issues, albeit with different numbers in two instances.[1] For the sake of clarity, in considering these four late-filed contentions we have renumbered them to continue the numbering sequence begun with the already-rejected environmental contentions. And below, we discuss the admissibility of each, beginning with renumbered contention EC-6.

1. CONTENTION EC-[6]: Environmental Impact Statement Required

> In the Environmental Assessment ("EA") for CP&L's December 23, 1998, license amendment application, the NRC Staff concludes that the proposed expansion of spent fuel storage capacity at the Shearon Harris nuclear power plant will not have a significant effect on the quality of the human environment. Environmental Assessment and Finding of No Significant Impact Related to Expanding the Spent Fuel Pool Stage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 10 (December 15, 2000). Therefore, the Staff has decided not to prepare an Environmental Impact Statement ("EIS") for the

[1] Originally-filed contention EC-2 corresponds to late-filed contention EC-1 and the previously submitted contention EC-5 corresponds to late-filed contention EC-4.

proposed license amendment. The Staff's decision not to prepare an EIS violates the National Environmental Policy Act ("NEPA") and NRC's implementing regulations, because the Finding of No Significant Impact ("FONSI") is erroneous and arbitrary and capricious. In fact, the proposed expansion of spent fuel pool storage capacity at Harris would create accident risks that are significantly in excess of the risks identified in the EA, and significantly in excess of accident risks previously evaluated by the NRC Staff in the EIS for the Harris operating license. These accident risks would significantly affect the quality of the human environment, and therefore must be addressed in an EIS.

There are two respects in which the proposed license amendment would significantly increase the risk of an accident at Harris:

(1) CP&L proposes several substantial changes in the physical characteristics and mode of operation of the Harris plant. The effects of these changes on the accident risk posed by the Harris plant have not been accounted for in the Staff's EA. The changes would significantly increase, above present levels, the probability and consequences of potential accidents at the Harris plant.

(2) During the period since the publication in 1979 of NUREG-0575, the NRC's Generic Environmental Impact Statement ("GEIS") on spent fuel storage[1], new information has become available regarding the risks of storing spent fuel in pools. This information shows that the proposed license amendment would significantly increase the probability and consequences of potential accidents at the Harris plant, above the levels indicated in the GEIS, the 1983 EIS for the Harris operating license, and the EA. The new information is not addressed in the EA or the 1983 EIS for the Harris operating license.

Accordingly, the Staff must prepare an EIS that fully considers the environmental impacts of the proposed license amendment, including its effects on the probability and consequences of accidents at the Harris plant. As required by NEPA and Commission policy, the EIS should also examine the costs and benefits of the proposed action in comparison to various alternatives, including Severe Accident Mitigation Design Alternatives ("SAMDAs") and the alternative of dry storage.

[1] NUREG-0575, Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (August 1979) (hereinafter "GEIS").

BCOC Contentions Request at 1-2.

DISCUSSION: *Id.* at 1-16; CP&L Contentions Response at 7-20; Staff Contentions Response at 7-26; BCOC Contentions Reply at 8-19.

RULING: With this contention, BCOC challenges the Staff's EA conclusion that the proposed CP&L license amendment to use spent fuel pools C and D does not require a complete EIS. In assessing the basis for this contention, we note that all three parties agree that the standard for requiring that an EIS be prepared is whether the action at issue, in this case the CP&L license amendment, is a major federal action having a significant impact on the human environment. *See* BCOC Contentions Request at 3; CP&L Contentions Response at 3 n.3; Staff Contentions Response at 8. Further, all the parties agree that the agency is not required to address in an EIS consequences of an action that are "remote

and speculative." *See* CP&L Contentions Response at 9-10; Staff Contentions Response at 16; BCOC Contentions Reply at 8. What the parties disagree about is whether a possible consequence of the action identified by BCOC — a severe accident in spent fuel pools C and D — is remote and speculative.

BCOC discusses a number of different elements that it asserts provide the basis for this contention, including the fact that the number of stored spent fuel assemblies at the Harris facility ultimately may double as a result of the proposed amendment; the purported impact of the use of "administrative measures" such as controlling fuel burnup levels rather than relying solely on "physical measures" such as fuel assembly separation and the presence of solid neutron absorbers to avoid criticality; and new information regarding sabotage risk. In the Board's view, however, the crux of the contention, and the focus of our consideration as to whether it meets the specificity and basis requirements of section 2.714, is whether the accident proposed by BCOC in basis F.1 of the contention has a probability sufficient to provide the beyond-remote-and-speculative "trigger" that is needed to compel preparation of an EIS relative to this proposed licensing action.

To examine whether the contention provides an adequate basis to support further Board consideration of this question, we examine the accident scenario in question, which was first summarized by CP&L, *see* CP&L Contentions Response at 9-10, with an appropriate modification by BCOC, *see* BCOC Contentions Reply at 8. In this regard, BCOC postulates the following chain of events:

(1) a degraded core accident;
(2) containment failure or bypass;
(3) loss of all spent fuel cooling and makeup systems;
(4) extreme radiation doses precluding personnel access;
(5) inability to restart any pool cooling or makeup systems due to extreme radiation doses;
(6) loss of most or all pool water through evaporation; and
(7) initiation of an exothermic oxidation reaction in pools C and D.

Relative to this accident sequence, what BCOC asserts, and what the CP&L and the Staff contest, is that BCOC has established an adequate basis to allow merits litigation on whether this sequence is not "remote and speculative" so that a further environmental analysis of the CP&L pool expansion amendment request is required.

In considering this question, we note that the Commission has provided some guidance regarding such an issue statement in its decision in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990). In that case, which also involved the expansion of a spent fuel pool, likewise at issue was the admission of a contention that asserted the license amendment involved required the preparation of an environmental impact statement because the action raised the potential for a substantial release of

radioactive material following the occurrence of a specific accident sequence. More specifically, the question in dispute was whether the accident sequence specified was of a sufficiently high probability to put it beyond the "remote and speculative" threshold for the purpose of admitting the contention.

Prior to coming before the Commission, however, that contention was considered by both the Licensing Board and the Appeal Board, with the matter coming before the Appeal Board on referral from the Licensing Board's admission of the contention. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29 (1989). The Appeal Board determined that:

> The essence of Environmental Contention 1 . . . is that an environmental impact statement is required for the proposed license amendment to assess the risks of the following hypothetical accident scenario: (1) a severe reactor accident occurs by some unidentified mechanism and involves substantial fuel damage, hydrogen generation, Mark I containment failure, and subsequent detonation in the reactor building where the Vermont Yankee fuel pool is located; (2) the reactor building and the spent fuel pool are assuredly not likely to withstand the pressure and temperature loads generated by such an accident, thereby threatening the pool cooling systems or the pool structure itself . . . ; and (3) pool heatup occurs, resulting in a self-sustaining zircaloy cladding fire with increased long-term health effects for the public from the increased fuel pool inventory.

Id. at 43 (citations omitted). The Appeal Board then went on to say that the scenario on which the contention is premised "is obviously not a 'normal' operating event; indeed it can be fairly characterized as a double 'worst case' accident." *Id.* Consequently, after what it considered to be a careful examination of the bases presented for the accident scenario, the Appeal Board rejected the contention and referred its ruling to the Commission. *See id.* at 52.

The Commission responded by remanding the issue to the Appeal Board for further consideration, saying:

> The Commission believes that on remand more information on the plausibility or probability of the reactor accident/hydrogen combustion/spent fuel pool cooling failure/cladding fire at issue here . . . is needed before a judgment should be made whether the accident . . . is remote and speculative. As part of our remand we therefore direct the Appeal Board to develop such information further. We leave it to the Appeal Board to decide on the procedural means to obtain this information, whether by inviting something akin to summary disposition motions or otherwise. If the Appeal Board finds that an accident probability on the order of 10^{-4} per reactor year is appropriate for the entire accident sequence postulated in this contention, the case should be returned to the Commission for further review. Otherwise, the Appeal Board should modify or confirm its judgment as to the remote and speculative nature of the accident on the basis of the accident probability derived on remand.

Vermont Yankee, CLI-90-4, 31 NRC at 335-36 (citations omitted).

There followed an Appeal Board request for clarification of the Commission's decision. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-938, 32 NRC 154 (1990). But before the Commission could respond, the intervenors asked to withdraw from the proceeding and the licensee moved to dismiss the proceeding. The Commission granted the motion to dismiss, but opined that it was

> concerned that the probability that the Appeal Board found to be so low as to be remote and speculative pertained not to the whole scenario in the contention but to pieces of the scenario in the contention or related scenarios set out in the technical documents, some with probabilities as high as on the order of 10^{-4} per reactor year. In ALAB-919, the Appeal Board bridged the gap between the technical documents and the scenario in the contention by assuming, conservatively, that the probability of that scenario could be no *greater* than certain scenarios actually analyzed in the documents. If the scenarios in the documents were remote and speculative, then, *a fortiori*, the scenario in the contention must be remote and speculative as well. Our opinion makes clear that future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual probabilities and accident scenarios being analyzed.

Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 132 (footnote omitted).

Certainly, in the intervening decade the Commission has come to rely on probabilistic analysis ever more heavily in the process of making decisions. Indeed, the entire trend in licensing, enforcement, inspection, and the granting of amendments has swung gradually toward decision-making by probabilistic risk assessment. We therefore think that the Commission's intent is at present even more firmly directed to deciding what is "remote and speculative" by examining the probabilities inherent in a proposed accident scenario.

In this instance, based on the information now presented by BCOC, including the 1993 Harris facility individual plant evaluation (IPE) of core damage frequency, the accident scenario it has postulated may have a probability in the range of 1×10^{-5} per reactor year, *see* BCOC Contentions Reply at 11-12, a figure that under the Commission's guidance seemingly should not be dismissed automatically as per se "remote and speculative." To be sure, CP&L and the Staff dispute various aspects of the BCOC probability analysis and its underlying accident scenario, including whether cooling water restoration would be precluded by onsite radiation levels; the availability of water makeup systems; bounding decay heat levels for pools C and D; the age of the spent fuel that will be stored in pools C and D; whether the probability of a substantial SFP release is on a par with the probability of a substantial reactor release; the effect of the use of burnup credit; and an increase in sabotage-related risk. And we agree with CP&L and the Staff that BCOC's assertions regarding sabotage risk do not provide a litigable basis for this contention. *See Philadelphia Electric Co.* (Limerick Generating Station, Units 1 and 2), ALAB-819, 22 NRC 681, 701 (1985), *review declined,*

CLI-86-5, 23 NRC 125 (1986), *aff'd, Limerick Ecology Action v. NRC,* 869 F.2d 719, 744 (3d Cir. 1989). We find, however, that the information provided by BCOC otherwise is sufficient to establish a genuine material dispute of fact or law adequate to warrant further inquiry relative to the other aspects of the BCOC scenario and the associated probability analysis.[2] Accordingly, we admit contention EC-6 as it relates to this accident sequence.[3]

Finally, in connection with further litigation on this contention, we offer the following additional observations. In its *Vermont Yankee* decision, the Commission directed the Appeal Board to select a "procedural means" to obtain the risk-informed information and suggested "something akin" to inviting summary disposition motions. CLI-90-4, 31 NRC at 336. In this instance, pursuant to 10 C.F.R. § 2.1109, CP&L has invoked the process set forth in Subpart K to Part 2 that includes the written summaries and oral argument specified in sections 2.1109 and 2.1113. Certainly, these procedures are sufficiently "akin" to summary disposition to satisfy the Commission's previously stated preference.

Additionally, so that we will be able properly to assess the significance of the materials submitted in the detailed written summaries required by section 2.1113(a), we ask that the parties address the following points:

1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on p. 95, *supra*? The estimates should utilize plant-specific data where available and should utilize the best available generic data where generic data are relied upon.

2. The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the ACRS's April 13, 2000 letter suggest that the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an exothermic reaction, greater than was assumed in the decade-old NUREG-1353)?

3. Assuming the Board should decide that the probability involved is of sufficient moment so as not to permit the postulated accident sequence to be classified as "remote and speculative," what would be the overall scope of the environmental impact analysis the Staff would be required to prepare (i.e, limited to the impacts of that accident sequence or a full blown EIS regarding the amendment request)?

[2] In this regard, we note that in our decision in LBP-00-12, 51 NRC at 259-60, in ruling on the two admitted BCOC technical contentions, we found CP&L's planned use of so-called "administrative processes," such as use of enrichment/burnup level controls and soluble boron as SFP criticality control measures, is permitted under General Design Criteria (GDC) 62. As a consequence, contrary to BCOC's assertion, the use of such measures does not, in and of itself, trigger the need for an EIS. Whether, and to what extent, the use of these control measures has any relevance to the probability calculation at issue here is a matter for resolution as part of further litigation regarding contention EC-6. The same is true for the question of the heat load for pools C and D, which seemingly includes an associated legal issue concerning appropriate project segmentation relative to NEPA.

[3] In its final sentence, the contention includes a statement about what should be analyzed in an EIS. For the reasons stated below relative to contentions EC-7 and EC-8, we consider this aspect of the contention premature and do not admit it.

2. *Contention EC-[7]: EIS Should Consider Cumulative Impacts in Light of New Information*

The EA is deficient because it fails to acknowledge or evaluate the significant cumulative environmental risk posed by the operation of pools A, B, C, and D.

BCOC Contentions Request at 16.

DISCUSSION: *Id.* at 17-18; CP&L Contentions Response at 20-25; Staff Contentions Response at 26-27; BCOC Contentions Reply at 20-21.

RULING: We find this contention premature, given that there is still an outstanding question whether the Staff correctly concluded in its EA that no environmental impact statement is required. *See* LBP-99-25, 50 NRC at 39. If, in ruling on the merits of contention EC-6, we should determine that an EIS is necessary, then the proper scope of that EIS would become a matter in controversy based on the CP&L environmental report (assuming the Staff requires that one be prepared) and the EIS the Staff prepares.

3. *Contention EC-[8]: Scope of EIS Should Include Brunswick and Robinson Storage*

The EIS for the proposed license amendment should include within its scope the storage of spent fuel from the Brunswick and Robinson nuclear power plants.

BCOC Contentions Request at 18.

DISCUSSION: *Id.* at 18-19; CP&L Contentions Response at 25-28; Staff Contentions Response at 27-28; BCOC Contentions Reply at 21.

RULING: As with contention EC-7, we decline to admit this contention as premature.

4. *Contention EC-[9]: Discretionary EIS Warranted*

Even if the Licensing Board determines that an EIS is not required under NEPA and 10 C.F.R. § 51.20(a), the Board should nevertheless require an EIS as an exercise of its discretion, as permitted by 10 C.F.R. §§ 51.20(b)(14) and 51.22(b).

BCOC Contentions Request at 20.

DISCUSSION: *Id.* at 20-23; CP&L Contentions Response at 28-30; Staff Contentions Response at 28-29; BCOC Contentions Reply at 21-22.

RULING: We have carefully considered whether such a discretionary EIS is warranted and we see no reason to require an EIS if one is not required by the rules. We recognize that CP&L and the Staff assert that such a requirement is ultra vires for this Board. *See* CP&L Contentions Response at 28; Staff Contentions Response at 28. We, however, need not rule on that point. Suffice it to say that we find no "special circumstances" pursuant to sections 51.20(b)(14) and 51.22(b) that would warrant a discretionary EIS.

III. ADMINISTRATIVE MATTERS

As we previously noted, under 10 C.F.R. § 2.1109, CP&L has invoked the procedural provisions of Part 2, Subpart K, relative to the litigation of this proceeding. Accordingly, the schedule for utilizing the Subpart K procedures in connection with contention EC-5 is as follows:

Discovery Begins	Monday, August 21, 2000
Discovery Ends	Friday, October 20, 2000
Written Summaries Filed	Monday, November 20, 2000

The discovery limitations and guidelines set forth in our July 29, 1999 issuance shall apply.[4] *See* Licensing Board Memorandum and Order (Granting Request to Invoke 10 C.F.R. Part 2, Subpart K Procedures and Establishing Schedule) (July 29, 2000) at 3-4. Moreover, the Board will establish a date and location for conducting oral argument regarding the parties' written summaries in a subsequent order.

IV. CONCLUSION

With these new proposed environmental contentions being filed within 45 days of the challenged Staff EA, the five-factor balancing test set forth in 10 C.F.R. § 2.714(a)(1) favors the admission of BCOC renumbered late-filed contentions EC-6 through EC-8. Additionally, we find that BCOC has established relative to contention EC-6 regarding "remote and speculative" SFP accident sequences that there exists a genuine material dispute of fact or law adequate to warrant further inquiry. We thus admit contention EC-6 and establish a schedule for its further litigation under 10 C.F.R. Part 2, Subpart K. On the other hand, we dismiss contentions EC-7 and EC-8, which concern the scope of any Staff EIS that may be needed, as premature, and dismiss contention EC-9, which concerns the need for a discretionary EIS, as lacking adequate support to show there exists a genuine material dispute of fact or law adequate to warrant further inquiry.

For the foregoing reasons, it is, this seventh day of August 2000, ORDERED that:

1. The following BCOC contention is *admitted* for litigation in this proceeding: EC-6.
2. The following BCOC contentions are rejected as inadmissible for litigation in this proceeding: EC-7, EC-8, and EC-9.
3. The parties are to conduct discovery and submit section 2.1113 written presentations in accordance with the schedule established in section III above.

THE ATOMIC SAFETY AND
LICENSING BOARD[5]

G. Paul Bollwerk, III
ADMINISTRATIVE JUDGE

Frederick J. Shon
ADMINISTRATIVE JUDGE

Dr. Peter S. Lam
ADMINISTRATIVE JUDGE

Rockville, Maryland
August 7, 2000

[4] As with the admitted technical contentions, the Board is not requiring that informal discovery must be used during the discovery period. Nonetheless, the Board notes that the parties need not await the beginning of the discovery period to initiate discussions regarding the nature and scope of the information each will be seeking in discovery and try to reach some agreement on documentary or other materials that can be provided without a formal discovery request.

Also, in connection with discovery in this proceeding, the Board notes that any attempt to obtain discovery materials or testimony from ACRS members, Staff, or consultants is subject to the exceptional circumstances showing of 10 C.F.R. § 2.720(h). *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-519, 9 NRC 42, 43 n.2 (1979). Moreover, the Board directs that any discovery requests regarding ACRS information or personnel must be filed within the first *10 days* of the discovery period established above.

[5] Copies of this Memorandum and Order were sent this date by Internet e-mail transmission to counsel for (1) Applicant CP&L; (2) Intervenor BCOC; and (3) the Staff.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Frederick J. Shon
Dr. Peter S. Lam

In the Matter of

Docket No. 50-400-LA
(ASLBP No. 99-762-02-LA)

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant)

May 5, 2000

In this 10 C.F.R. Part 2, Subpart K spent fuel pool (SFP) expansion proceeding, in accordance with 10 C.F.R. § 2.1115, the Licensing Board denies the request of Intervenor Board of Commissioners of Orange County, North Carolina (BCOC), to designate for an evidentiary hearing either of the two admitted technical contentions regarding SFP criticality and SFP cooling system piping quality assurance. The Board concludes that (1) BCOC has failed to show there is a genuine and substantial dispute of fact or law that can be resolved only by the introduction of evidence at an evidentiary hearing; and (2) Applicant Carolina Power and Light Company (CP&L) has met its burden to establish that its proposed licensing action is in compliance with the requirements of the Atomic Energy Act and the agency's implementing regulations.

RULES OF PRACTICE: BURDEN OF PROOF (SUBPART K PROCEEDING)

Notwithstanding the agency's rules of practice that place the ultimate burden of proof on the license applicant with respect to a merits disposition of any

substantive matter at issue, relative to the central 10 C.F.R. Part 2, Subpart K issue of the existence of disputed material facts requiring an evidentiary hearing, "the burden . . . [is] on the party requesting adjudication." 50 Fed. Reg. 41,662, 41,667 (1985) (statement of considerations for final rule adopting 10 C.F.R. Part 2, Subpart K).

REGULATORY CONSTRUCTION OR INTERPRETATION: REGULATORY HISTORY

When regulatory language is ambiguous, it is appropriate to resort to the regulatory history of the provision to see what light, if any, it sheds on the question. *See Kansas Gas & Electric Co.* (Wolf Creek Generating Station, Unit 1), CLI-99-19, 49 NRC 441, 456 (1999) (ambiguity in statutory language requires resort to legislative history).

RULES OF PRACTICE: DISCOVERY (AVAILABILITY IN SUBPART K PROCEEDING AFTER CONTENTION DESIGNATED FOR EVIDENTIARY HEARING)

Nothing in 10 C.F.R. Part 2, Subpart K, suggests that additional discovery is available if an evidentiary hearing is found to be necessary in accordance with 10 C.F.R. § 2.1115.

RULES OF PRACTICE: NONTIMELY SUBMISSION OF CONTENTIONS (REQUIREMENT TO ADDRESS LATE-FILING STANDARDS)

Because a claim not part of an admitted contention can only be considered if it fulfills the 10 C.F.R. § 2.714(a) late-filing standards, a petitioner's failure to address those standards precludes further consideration of the issue. *See Boston Edison Co.* (Pilgrim Nuclear Power Station), ALAB-816, 22 NRC 461, 465-68 (1985).

TECHNICAL ISSUES DISCUSSED

The following technical issues are discussed: quality assurance; spent fuel pool cooling systems; spent fuel pool criticality.

MEMORANDUM AND ORDER

(Ruling on Designation of Issues for an Evidentiary Hearing)

Pending before the Licensing Board in this 10 C.F.R. Part 2, Subpart K pro ceeding are the parties' pleadings addressing the question whether, in accordanc with 10 C.F.R. § 2.1115, to designate for an evidentiary hearing either of the tw admitted issues of Intervenor Board of Commissioners of Orange County, Nort Carolina (BCOC). With these contentions — Technical Contention 2 (TC-2), In adequate Criticality Prevention, and Technical Contention 3 (TC-3), Inadequat Quality Assurance — BCOC challenges Carolina Power and Light Company' (CP&L) December 23, 1998 application to amend the operating license for it Shearon Harris Nuclear Power Plant (Harris or HNP) to permit the addition of rac modules to spent fuel pools (SFPs) C and D and to place those pools in service BCOC asserts that it has established there are disputed material facts relative t each of the contentions that warrant further exploration in an evidentiary hearing In contrast, CP&L and the NRC Staff declare that BCOC has failed to establis there is any need for such an additional proceeding and,, as a consequence, th portion of this proceeding relating to these contentions should be dismissed.

For the reasons set forth below, we find relative to the issues raised b contentions TC-2 and TC-3 that (1) BCOC has failed to show there is a genuin and substantial dispute of fact or law that can be resolved only by the introductio of evidence at an evidentiary hearing; and (2) based on the record before us Applicant CP&L has met its burden to establish that its proposed licensing actio is in compliance with the requirements of the Atomic Energy Act and the agency' implementing regulations, warranting disposition of these issues in its favor.

I. BACKGROUND

A. Procedural Matters

In the Board's ruling in LBP-99-25, 50 NRC 25 (1999), in which we found tha Intervenor BCOC had standing and had presented admissible contentions so as t warrant its admission as a party to this proceeding, we described the circumstance surrounding the CP&L license amendment request as follows:

> In its December 1998 license amendment request, CP&L indicated that the fuel handling building (FHB) at the Harris site was originally designed and constructed with four separate spent fuel pools to accommodate the four reactor units that were planned for the site. Pools A through D were anticipated to serve Units 1 through 4, respectively. Although three of the units were canceled in the early 1980s, the FHB, the four pools (with liners), and the cooling and cleanup system to support pools A and B were completed and turned over to CP&L. Construction on the cooling and cleanup system for pools C and D, however, was

not completed. CP&L also declared that because a Department of Energy high-level waste repository is not expected to be available in the foreseeable future, it has been shipping spent fuel from its three other nuclear facilities for storage in the Harris pools in order to maintain full core offload capability for those facilities. According to CP&L, the present amendment request to utilize pools C and D is designed to provide storage capacity for all four CP&L units — Harris, Brunswick Steam Electric Plant, Units 1 and 2, and H.B. Robinson, Unit 2 — through the end of their current operating licenses.

Id. at 27-28 (citation omitted). Relative to the CP&L amendment request, we admitted contentions TC-2 and TC-3. As admitted, TC-2 provides:

> 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. In doing so, we found that this contention was adequately supported by two bases, which were summarized as follows:

> Basis 1 — CP&L's proposed use of credit for burnup to prevent criticality in pools C and D is unlawful because GDC 62 prohibits the use of administrative measures, and the use of credit for burnup is an administrative measure.
>
> * * * *
>
> Basis 2 — The use of credit for burnup is proscribed because Regulatory Guide 1.13 requires that criticality not occur without two independent failures, and one failure, misplacement of a fuel assembly, could cause criticality if credit for burnup is used.

Id. at 35, 36. So too, we admitted contention TC-3, which provides:

> 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 the 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 Alternative 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 the remote camera, and the weld quality cannot be demonstrated adequately by circumstantial evidence.

Id. at 36-37.

Following the Board's ruling on standing and contentions, as was its right pursuant to 10 C.F.R. § 2.1109, Applicant CP&L filed a timely request that the procedural construct of 10 C.F.R. Part 2, Subpart K, be utilized to conduct this proceeding. As a consequence, in accordance with section 2.1111, the Board gave the parties a limited period within which to conduct discovery regarding these contentions.[1] Thereafter, as is provided for in section 2.1113(a), on January 4, 2000, the parties submitted written summaries of the facts, data, and arguments on which they intended to rely at an oral argument intended to provide them with an opportunity to discuss whether or not there were any genuine and substantial factual or legal disputes that merited further exploration in an evidentiary hearing. Then, on January 21, 2000,[2] the Board conducted a day-long proceeding in which it entertained the parties' oral presentations on the question whether there were disputed factual or legal issues relative to either of the admitted contentions that merited further consideration in an evidentiary hearing with live witnesses and party cross-examination. *See* Tr. at 190-442.

B. Technical/Regulatory Matters

As it is relevant to this proceeding and is described in the NRC Staff's January 4, 2000 written summary, the December 23, 1998 CP&L license amendment request at issue in this proceeding contains two parts:[3]

> 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

[1] Although the original discovery period was limited to the 90 days specified in section 2.1111, the Board granted an unopposed 4-day extension to permit several depositions to be completed. *See* Licensing Board Order (Granting Discovery Extension Request) (Oct. 18, 1999) at 1-2 (unpublished).

[2] Although section 2.1113(a) provides that the parties' oral presentations should occur within 15 days of the filing of the parties' written summaries, the 17-day interval here was arrived at after consultation with the parties in response to a Staff request to extend the deadline originally set for the filing of written summaries. *See* Licensing Board Memorandum and Order (Extending Time for Written Summaries and Oral Argument) (Dec. 13, 1999) at 1- (unpublished).

[3] The application also addresses a safety issue regarding the additional heat load on the component cooling water system, which is not part of the current controversy before the Board. *See* NRC Staff Brief and Summary of Relevant Facts, Data and Arguments upon Which the Staff Proposes to Rely at Oral Argument on Technical Contentions 2 and 3 (Jan. 4, 2000) at 7.

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 [K-effective (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 [CCW system] 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 [American Society of Mechanical Engineers (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 provide[] an acceptable level of quality and safety.

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 (Jan. 4, 2000) at 5-6 [hereinafter Staff Summary].

Relative to these revisions, the scope and interpretation of several regulatory provisions are at issue. In the case of contention TC-2, which concerns the issue of criticality control,[4] a measure of significant concern is General Design Criterion (GDC) 62, which 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.

[4] In its January 4, 2000 summary, the Staff provides the following discussion of criticality that outlines the basic technical principals involved relative to contention TC-2:

> Criticality is the achievement of a self-sustaining nuclear chain reaction. The chain reaction proceeds as atoms of a fissile material absorb slow (thermal) neutrons and split (fission) into new light atoms (i.e., fission products) and additional neutrons that, in turn, interact with additional fissile atoms. Neutrons resulting from fission have high energy and are called "fast" neutrons. Fast neutrons are not readily captured in U-235, the fissile material originally present in fresh fuel. Rather, a neutron must lose energy and "slow down," or become "thermalized" (a thermal neutron), in order to be readily captured in U-235 and cause fission.
>
> In order for fast neutrons to slow down, they must collide with, and transfer energy to, atoms. This process is called "moderation." A light element (such as hydrogen) is an effective moderator because the mass of its nucleus is on the same order as that of a neutron. Therefore, upon initial collision, the neutron imparts most of its energy to the hydrogen nucleus and becomes thermalized. Water, with its high hydrogen content, is the moderator in a light water reactor (LWR) such as Harris.
>
> After being created through fission, during the process of moderation, and after reaching thermal energy levels, a neutron may undergo several events. It may be absorbed by nonproductive capture in the fuel, the moderator, or the structural materials. It may leak from the reactor system and either be reflected back into the system or be lost. Finally, it may be absorbed by the U-235, cause fission, and produce more fast neutrons.

(Continued)

10 C.F.R. Part 50, App. A, Criterion 62. Also at issue is the so-called "double contingency principle" (DCP) of Staff draft Regulatory Guide 1.13, App. A, at 1.13-9 (proposed rev. 2, Dec. 1981) (emphasis in original), which states:

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

In connection with contention TC-3 and the so-called Alternative Plan submitted by CP&L to show that its cooling and cleanup system piping meets agency regulatory requirements, several different provisions of 10 C.F.R. § 50.55a are potentially relevant, including the following:

> (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.
>
> * * * *
>
> (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 that:
>
> (i) The proposed alternative would provide an acceptable level of quality and safety . . . .

In particular, BCOC contends that the CP&L Alternative Plan proposal fails to satisfy three of the quality assurance (QA) criteria of Appendix B to 10 C.F.R. Part 50. In describing these criteria, the Staff correctly notes:

> When the process continues on its own, the system of atoms of fissile material is said to be critical. The measure of criticality is the effective neutron multiplication factor, *k*-effective, or k_{eff}. The multiplication factor is the ratio of the rate of neutron production to neutron loss due to fission, nonproductive capture, and leakage. *K*-infinity, or k_{inf}, is the infinite multiplication factor, which refers to the neutron multiplication of an infinite system. 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. Criticality is achieved when k_{eff} is equal to 1.0. When k_{eff} is less than 1.0, the system is subcritical. 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. No array of LWR fuel can achieve criticality without water moderation present in the array. Well-developed mathematical models (equations) exist in present-day computer codes and are used to compute k_{eff}.
>
> "Reactivity" is defined as $(k_{eff} - 1)/k_{eff}$. When fuel is irradiated in a reactor as a result of operation and power generation, the reactivity of the fuel decreases over the design life of the fuel assembly. This reduction of reactivity with irradiation is called "burnup." Burnup is caused by the change in fissile content of the fuel (i.e., depletion of U-235 and production of Pu-239 and other fissile actinides), the production of actinide absorbers, and the production of fission product neutron absorbers. Before each reactor operating cycle, a licensee performs a reload analysis that predicts the burnup of each fuel assembly during the cycle. These calculations are confirmed during the cycle by measurements of various operating characteristics, such as boron concentration and power distribution. After every operating cycle (typically 1 to 2 years), approximately 1/3 of the fuel in a reactor is removed because its reactivity is too low to effectively contribute to power generation in the reactor environment. This irradiated (or spent) fuel is generally placed in a spent fuel pool at the reactor site and is replaced in the reactor by fresh (unirradiated) fuel.

Staff Summary at 20-22 (citations omitted).

Appendix B requires the development and application of a [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 man[a]gerial 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.["]

Staff Summary at 11.

II. ANALYSIS

A. Standards Governing 10 C.F.R. § 2.1115 Determination Regarding the Need for an Evidentiary Hearing to Resolve Admitted Issues

The procedures in 10 C.F.R. Part 2, Subpart K, were established in response to a congressional mandate found in the Nuclear Waste Policy Act of 1982 (NWPA). Specifically, NWPA § 134(a)-(b), 42 U.S.C. § 10154(a)-(b), states that for any reactor operating license amendment "to expand the spent nuclear fuel storage capacity at the site of a civilian nuclear power reactor," the Commission is to provide parties to any hearing proceeding on the expansion amendment with the opportunity to present facts, data, and arguments, by way of written summaries and sworn testimony, and an oral argument. Based on the summaries and the argument, the Commission then is to designate "any disputed questions of fact, together with any remaining questions of law, for resolution in an adjudicatory hearing" if the Commission finds that "there is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence at an adjudicatory hearing," and "the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute." Sections 2.1113 and 2.1115 of 10 C.F.R. incorporate these requirements. In addition, section 2.1115(a)(1)-(2) provides that the presiding officer shall "[d]esignate any disputed issues of fact, together with any remaining issues of law, for resolution in an adjudicatory hearing," and "[d]ispose of any issues of law or fact not designated for resolution in an adjudicatory hearing." Moreover, as we have previously noted, notwithstanding the agency's rules of practice that place the ultimate burden of proof on CP&L, as the license Applicant, with respect to a merits disposition

of any substantive matter at issue in this proceeding (i.e., the admitted BCOC contentions), relative to the central Subpart K issue of the existence of disputed material facts requiring an evidentiary hearing, " 'the burden . . . [is] on the party requesting adjudication.' " Licensing Board Memorandum and Order (Subpart K Oral Argument Procedures) (Jan. 13, 2000) at 2 (quoting 50 Fed. Reg. 41,662, 41,667 (1985) (statement of considerations for final rule adopting 10 C.F.R. Part 2, Subpart K)) (unpublished).

It is against these standards that we review the parties' filings and oral argument presentations.

B. Contention TC-2

1. Basis One

DISCUSSION: Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which [BCOC] Intends to Reply at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with [CP&L] Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with Respect to Criticality Prevention Issues (Contention TC-2) (Jan. 4, 2000) at 19-41 [hereinafter BCOC TC-2 Summary]; Summary of Facts, Data, and Arguments on Which Applicant Proposes to Rely at the Subpart K Oral Argument (Jan. 4, 2000) at 29-55 [hereinafter CP&L Summary]; Staff Summary at 31-40; Tr. at 218-32, 254-62, 276-78, 285-86, 287-92, 296-98, 305-08.

a. BCOC Position

Regarding basis one of contention TC-2, referencing the supporting affidavit of Institute for Resource and Security Studies executive director Dr. Gordon Thompson, BCOC asserts that the CP&L license application is inadequate because it places impermissible reliance on administrative procedures and controls for criticality prevention. Instead, according to BCOC, CP&L should be relying entirely on physical systems or processes as required by the proper interpretation of GDC 62. Noting that under Part 50, Appendix A, GDCs are considered principal reactor design criteria minimum requirements, *see* 10 C.F.R. Part 50, App. A, Introduction, BCOC declares that the requirement of GDC 62 that criticality in a facility's fuel storage and handling system must be prevented by "physical systems or processes, preferably by use of geometrically safe configurations," clearly precludes the use of administrative controls, such as the burnup/enrichment level controls and SFP soluble boron presence that are being relied upon by CP&L to avoid criticality problems. According to BCOC, this follows from the plain language of GDC 62, which specifies physical systems or processes and provides the example of safe fuel bundle geometrical configurations. Moreover,

BCOC declares, notwithstanding the fact that any physical measure has some administrative component, there is a basic difference between a physical and administrative measure in that the latter requires continuing human interaction and concomitantly is subject to human error.

Relative to the first point, BCOC asserts that the rulemaking history of GDC 62 supports its plain language argument, including Atomic Energy Commission (AEC) pre-rulemaking documents; the June 1967 AEC draft GDC, which (like the pre-rulemaking documents) stated that "[s]uch means as geometrically safe configurations shall be emphasized over procedural controls"; September 1967 Oak Ridge National Laboratory (ORNL) comments on the draft criticizing the reference to procedural controls; and the AEC February 1971 final rule, which provided the present GDC 62 language without any reference to procedural controls. BCOC further maintains that other relevant NRC criticality standards, including (1) 10 C.F.R. § 70.24, regarding criticality monitoring for significant special nuclear material quantities; (2) section 50.68, which establishes a blanket exemption from section 70.24 for those agreeing to follow specified criticality accident prevention requirements; and (3) section 72.124, which establishes criticality control measures for independent spent fuel storage installations (ISFSIs), do not contradict this plain language meaning.

Against this backdrop, BCOC concludes it is clear that the CP&L license amendment proposal to restrict the burnup/enrichment of the fuel being placed in the pools to suppress criticality, which relies on ongoing administrative controls to maintain those limits, violates the language and intent of GDC 62. Nor does Staff draft Regulatory Guide 1.13, which allows fuel enrichment and burnup limits for spent fuel pool criticality control, permit a different result given that this Staff guidance document cannot modify or circumvent a regulatory requirement like GDC 62. Finally, according to BCOC, the Staff's willingness to permit CP&L (and numerous others) to use burnup/enrichment controls under Regulatory Guide 1.13 without performing any kind of a systematic safety analysis is inconsistent with its public health and safety responsibilities, particularly in light of several reported incidents involving SFP assembly mispositioning and a boron dilution event that are described in Appendix C to the BCOC January 4, 2000 summary.

b. CP&L Position

CP&L first asserts that BCOC has impermissibly changed its position regarding basis one from the assertion that no administrative measures are allowed under GDC 62 to a declaration that there are appropriate administrative measures and that the burnup/enrichment controls sought by CP&L fall into the impermissible category because those measures must be maintained on an ongoing basis. Additionally, referencing the affidavit of Holtec International Senior Vice President and Chief Nuclear Scientist Dr. Stanley E. Turner, CP&L declares that the Staff's consistent interpretation of GDC 62 to allow burnup/enrichment limits is appropriate because (1) every practical spent fuel pool criticality control measure — geometric separation, solid neutron absorbers, soluble neutron absorbers, fuel reactivity, and fuel burnup — is a physical process or system involving some administrative measures; (2) the regulatory history of GDC 62 shows that administrative measures have always been understood to be part of criticality control physical systems or processes; (3) the recently adopted section 50.68 explicitly contemplates and permits criticality control administrative measures, including fuel enrichment and burnup limits; (4) the Staff's two-decades-old interpretation of GDC 62 should be accorded considerable weight; and (5) the new BCOC interpretation highlights the absurdity of its original, "no administrative measures" position.

On the initial point, CP&L declares that BCOC has admitted in discovery that the five criticality control measures listed above are physical systems or processes and that each is implemented using administrative measures. CP&L also maintains there is nothing in GDC 62 that differentiates between criticality controls based on the timing or duration of the implementing administrative measures involved. In connection with the regulatory history of GDC 62, CP&L maintains that the metamorphosis from the July 1967 draft standard referenced by BCOC to the final language establishes that the reference to "physical procedures or processes" includes administrative controls like enrichment/burnup credits while the stated preference for the "use of geometrically safe configurations" is not intended to foreclose the use of such administrative controls.

Regarding 10 C.F.R. § 50.68, which provides requirements intended to prevent criticality accidents in instances when a section 70.24 monitoring system is not utilized, CP&L asserts that this recently adopted provision also establishes the viability of administrative controls under GDC 62. Noting that, like GDC 62, section 50.68 is intended to prevent inadvertent criticality events, CP&L discusses various Staff and Commission statements in the context of the 1998 rulemaking regarding section 50.68 that it contends establish these administrative controls are permissible under GDC 62. CP&L also relies on the language of section 50.68(b)(4) regarding the effects of fuel burnup, which it finds implies the fuel burnup limits are a permitted criticality control method, and of section 50.68(b)(7) permitting the use of fuel enrichment limits for criticality control, as evidence that these control measures are within the confines of GDC 62.

Also compelling, CP&L declares, is the consistent Staff interpretation of GDC 62 to include the use of fuel enrichment and burnup limits for criticality control, which goes back to the adoption of draft Regulatory Guide 1.13 in 1981, and includes some twenty Staff license amendment approvals of the use of fuel enrichment and burnup limits as criticality controls. Also relevant, CP&L asserts, is the Staff's August 1998 Criticality Guidance document, which CP&L declares effectively replaces Regulatory Guide 1.13 and approves fuel enrichment and burnup limits as criticality control measures.

Finally, CP&L disparages what it labels BCOC's attempt to change its admitted contention during discovery by outlining a position that some administrative measures are permitted under GDC 62, but not those proposed by CP&L relative to its SFP expansion request. In addition to being impermissibly late, CP&L asserts, there is nothing in the text of GDC 62 that differentiates between criticality control methods based on the timing and duration of administrative measures implementation. It also finds inapposite the BCOC Summary Appendix C incidents involving SFP assembly mispositioning and a boron dilution event. According to CP&L, of the nineteen incidents specified, only six apparently involve fuel misplacement, as would be relevant to the BCOC contention, and of those, five involve fuel loading in a checkerboard pattern that is not applicable to the Harris facility. The sixth, involving a failure to verify independently fuel move sheets, also is not applicable, according to CP&L, because, as is explained in the accompanying affidavit of CP&L Spent Pool Project Supervisor R. Steven Edwards, CP&L has a series of redundant checks that will prevent such an incident from occurring.

c. Staff Position

According to the Staff, the language of GDC 62, its regulatory history, Staff practice under that provision, and agency adjudicatory and rulemaking action authorizing the use of administrative controls to prevent criticality, all support the CP&L position relative to this portion of contention TC-2. Like CP&L, the Staff finds that the change in the language of GDC 62 from the original AEC proposal to the present wording does not preclude the use of administrative controls, but instead reflects a preference for geometrical configurations as a criticality control measure. The Staff also notes that because GDC 62 applies to both fuel handling and fuel storage systems and because the former necessarily requires the use of administrative controls as single fuel assemblies are moved, to adopt the BCOC reading of that provision would undermine the imposition of fuel handling criticality requirements. In addition, the Staff declares that over the past 18 years under GDC 62 it consistently has authorized the administratively controlled criticality measure of burnup credit without an accident, permissions that in several instances were subjected to unsuccessful adjudicatory challenges. Also, the Staff points out, several agency adjudicatory decisions appear to accept the Staff-endorsed concept of administrative controls to prevent SFP criticality, including *Consumers Power Co.* (Big Rock Point Nuclear Plant), ALAB-725, 17 NRC 562, 564-65, 571 (1983), and *Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 1), LBP-89-12, 29 NRC 441, 454-56, *aff'd on other grounds,* ALAB-921, 30 NRC 177 (1989). Moreover, according to the Staff, in adopting section 50.68 in 1998, the agency endorsed the use of administrative controls relative to the criticality control measure of soluble boron credit (section 50.68(b)(2)-(4)). Finally, the Staff rejects the BCOC assertion that the absence

of human actions and administrative controls makes dry cask storage safer than SFP storage as beyond the scope of the contention and not reflective of the Commission's determination that both storage methods are safe.

d. Board Ruling

Although BCOC declares the language of GDC 62 to be clear, we find it considerably less than so in the context of this dispute. As the shifting debate between the parties over the scope of the term "physical procedures or processes" illustrates, there is no clear-cut demarcation to differentiate the administrative and nonadministrative aspects of the criticality control procedure/processes at issue here so as to place any of them either inside or outside this label.[5] As such, we think it appropriate to resort to the regulatory history of this provision to see what light, if any, it sheds on the question of whether the enrichment/burnup/boron solubility measures proposed by CP&L fall within the confines of those criticality control measures sanctioned by GDC 62. *See Kansas Gas & Electric Co.* (Wolf Creek Generating Station, Unit 1), CLI-99-19, 49 NRC 441, 456 (1999) (ambiguity in statutory language requires resort to legislative history).

CP&L and the Staff have the better of the argument here. The critical item is the action of the AEC, the NRC's regulatory predecessor, in response to the comments of ORNL to the 1967 proposed rule version. At that juncture, the proposed GDC provided:

> 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.

CP&L Summary, exh. 16A (32 Fed. Reg. 10,213, 10,217 (1967)). From this formulation, it is clear that with the term "physical systems or processes," the first sentence defines the scope of the appropriate methods of criticality control, while the second expresses the agency's preference among those methods, i.e., geometries over other controls. In response to this proposed rule, the Commission received a comment from ORNL that expressed uncertainty over the implications of the reference to "processes" at the end of the first sentence and declared that "nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors." *Id.* exh. 17A (Sept. 6, 1967 Letter from William B. Cottrell, Director, ORNL Nuclear Safety information Center, to H.L. Price, AEC Director of Regulation, encl. at 11)). ORNL thus suggested that the last sentence be changed to read "'[s]uch

[5] Indeed, the fact that none of the parties seems to be able to define a criticality control procedure that falls wholly inside of or outside of the realm either of the "physical" or the "administrative" strengthens our resolve on this point. *See* Tr. at 226-28, 261-62.

means as geometrically safe configurations shall be used to insure that criticality cannot occur.' " *Id.* Albeit without discussion, the agency revised the final rule to its present configuration by incorporating the second suggestion, i.e., to indicate that geometric configuration is a preference, but without deleting the reference to "processes" or, it seems apparent, the administrative measures they encompass.

While this arguably is dispositive of the matter at issue in this portion of contention TC-2, we also agree with CP&L and the Staff that further support for this conclusion comes from recent agency adoption of section 50.68 and the longstanding Staff interpretation embodied in draft Regulatory Guide 1.13 and prior adjudicatory treatment of criticality-related matters. The language of section 50.68(b)(2), (4), (7) seems to contemplate the use of enrichment, burnup, and soluble boron as criticality control measures.[6] So too, the Staff's nearly 20-year-old interpretation in the context of draft Regulatory Guide 1.13, albeit not dispositive, nonetheless reinforces our conclusion that this is the appropriate construction of this provision, see *Petition for Emergency and Remedial Action*, CLI-78-6, 7 NRC 400, 406-07 (1978), as do the adjudicatory decisions cited by the Staff.

Finally, the "problem" cases discussed by BCOC in Appendix C to its written summary as evidence of the Staff need to require an additional analysis are wholly inadequate as a basis for further adjudicatory proceedings relative to this concern. As CP&L correctly notes, the fuel mispositioning cases are not relevant to the Harris configuration and, as is apparent from the discussion of boron control measures in the affidavit of Mr. Stevens, *see* CP&L Summary, exh. 1, at 15-17, the boron dilution incident cited by BCOC has little relevance in the context of the Harris facility.[7]

In sum, in accordance with 10 C.F.R. § 2.1115(a), we conclude relative to this portion of contention TC-2 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based upon the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

[6] In connection with this provision, we note that section 50.68(b)(4) uses the term "maximum fuel assembly reactivity." Although it does not affect our determination regarding this provision, we note that "reactivity" is generally considered to be a property of the entire SFP rather than an individual fuel assembly. Individual assemblies are considered to have "reactivity worth," a value influenced by parameters such as original enrichment, burnup, irradiation history, element design, and pool position, that is imparted to the pool's reactivity value upon insertion.

[7] When questioned about the seeming lack of "significance" of these incidents, BCOC's response was to promise more information after further discovery. *See* Tr. at 242; *see also* Tr. at 439-41. Putting aside the fact that nothing in Subpart K suggests that additional discovery is available if an evidentiary hearing is found to be necessary, this response is not one likely to provide an impetus for the Board to convene such a hearing.

2. *Basis Two*

DISCUSSION: BCOC TC-2 Summary at 41-46; CP&L Summary at 55-74; Staff Summary at 26-29, 40-44; Tr. at 232-39, 245-46, 262-85, 292-304, 308-15, 318-24.

a. *BCOC Position*

BCOC has provided four interrelated arguments regarding basis two of contention TC-2. Among other things, the BCOC summary is supported by the affidavit of Dr. Gordon Thompson and Appendix C to its January 4, 2000 summary, discussed above, that describes some incidents it believes are relevant to the potential for criticality in spent fuel pools.

Relative to this portion of contention TC-2, BCOC first asserts that draft Regulatory Guide 1.13 calls for the analysis of situations under the double contingency principle involving "at least" two failures or violations of operating limits. According to BCOC, for an analysis to meet this requirement, it must identify the sets of failures or violations that might cause criticality, and then evaluate these failures or violations in combinations of at least two, to determine which combinations will cause criticality. This process will yield an envelope of criticality that bounds the combinations of failures and violations that produce criticality. BCOC states that such an envelope cannot be identified if failures or violations are evaluated one at a time. When the envelope has been identified, the DCP can be applied, with consideration as to whether the failures or violations are unlikely, independent, or concurrent. BCOC argues that CP&L has not gone through this process, but has only considered a single failure, limited to the mispositioning of one fresh PWR fuel assembly.

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

BCOC further declares that, in considering possible criticality accidents at Harris pools C and D, CP&L assumes that the mispositioning of fuel is an unlikely event but CP&L offers no evidence to support this assumption. BCOC maintains that, as shown in Appendix B and discussed in Appendix C of its January 4, 2000 filing experience shows that fuel mispositioning is likely. Moreover, BCOC believes that in a criticality accident involving fuel mispositioning and soluble boron dilution

these events will typically be consecutive rather than concurrent. High-reactivity fuel could be mispositioned in a fuel pool prior to or after a boron dilution event, or during both periods if an event sequence involving mispositioning of multiple fuel assemblies spans a time period during which boron dilution occurs. BCOC argues that, were CP&L to treat fuel mispositioning as a likely occurrence, then the criticality analysis would necessarily consider fuel mispositioning in combination with a complete absence of soluble boron. Indeed, BCOC asserts, this would be so even employing the allegedly invalid, nonconservative version of the DCP that is articulated in the so-called Kopp Memorandum, an August 19, 1998 memorandum providing guidance on regulatory requirements relating to SFP criticality analysis authored by Staff witness Dr. Laurence Kopp, an NRC senior reactor engineer. Similarly, were CP&L to consider mispositioning and soluble boron dilution as consecutive occurrences, the criticality analysis would necessarily consider these occurrences in combination. BCOC states that calculations by CP&L and the Staff, summarized in Appendix C to its January 2000 summary, show that mispositioning of a single fresh PWR fuel assembly in Harris pools C or D would, in the absence of soluble boron, cause the *k*-effective to exceed the regulatory limit of 0.95. Therefore, BCOC believes that mispositioning of more than one assembly could result in a potentially serious supercritical configuration.

In addition, BCOC maintains that, in considering the role of fuel mispositioning as a potential cause of criticality, CP&L has limited its attention to the mispositioning of only one PWR fuel assembly. Underlying this restriction is an assumption that a single failure or violation will lead to the mispositioning of only one fuel assembly. BCOC asserts that Appendices B and C to its January 2000 summary demonstrate that a single error can lead to the mispositioning of multiple fuel assemblies. BCOC thus claims that, in addition to its improper reliance on administrative measures for criticality control, CP&L's misapplication of the DCP in the manner discussed above has yielded a criticality analysis that is nonconservative and inadequate to provide reasonable assurance that public health and safety will be protected in the event of an accident.

To support this position, referencing the December 1998 CP&L license amendment application, the affidavit of Staff witness Dr. Kopp, and an October 1983 American National Standards Institute (ANSI) Standard 57.2-1983, entitled "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," BCOC asserts that the *K*-effective value for SFP criticality must be less than 0.95, with a 95% probability at a 95% confidence level, under all conditions. More specifically, BCOC asserts that this requirement of keeping *K*-effective below 0.95 applies even under the scenario in which a fresh fuel assembly is misplaced concurrent with the accidental loss of all soluble boron. BCOC thus maintains that all the analyses CP&L and the Staff have performed and provided in their January 4, 2000 summaries only demonstrate that CP&L has not resolved

the factual dispute as to whether a single misplaced spent assembly would result in criticality above acceptable levels under applicable NRC and industry standards.

b. CP&L Position

CP&L supports its January 4, 2000 summary on this matter with the affidavits of Dr. Everett L. Redmond II, a nuclear engineer with Holtec International with responsibility for performing nuclear criticality analyses for spent fuel storage systems, and Michael J. DeVoe, a CP&L nuclear engineer responsible for performing the CP&L review of the nuclear criticality analyses for Harris spent fuel pools C and D. Regarding this portion of the contention, CP&L first asserts that basis two raises a question of fact, i.e., will a single misplaced fuel assembly, involving a fuel element of the wrong burnup or enrichment, cause criticality in Harris SFPs C and D? According to CP&L, disposition of this question requires the resolution of two additional queries: (1) Did CP&L perform a criticality analysis of a single fuel assembly misplacement, involving a fresh fuel assembly with the maximum permissible reactivity at Harris, for the spent fuel storage racks in Harris pool C and D; and (2) does that criticality analysis demonstrate that a single fuel assembly misplacement, involving a fresh fuel assembly with the maximum permissible reactivity at Harris, will not cause criticality in Harris pools C and D? CP&L claims the Board should dispose of basis two in its favor because these two questions can be answered in the affirmative.

CP&L declares that following the admission of basis two, it performed an analysis to evaluate the misplacement of a single fuel assembly in the spent fuel storage racks for Harris SFPs C and D. The results of this analysis are documented in Holtec Report No. HI-992283, Evaluation of Fresh Fuel Assembly Misload in Harris Pools C and D (rev. 0 Sept. 20, 1999), which CP&L refers to as the Harris Misplacement Analysis. The analysis, performed by Dr. Redmond, evaluates a fuel assembly misplacement specifically for the spent fuel storage racks for Harris pools C and D using the specific fuel assembly characteristics and spent fuel storage rack designs for Harris spent pools C and D. The analysis uses the same methodology, including the assumptions and modeling of the storage rack design and fuel assembly characteristics, as was developed for — and used in — the so-called Harris Base Criticality Analysis that was generated initially for the CP&L license amendment application.

According to CP&L, the misplacement analysis evaluates a single fresh fuel assembly mispositioning of the maximum permissible enrichment for Harris in a spent fuel storage rack that is otherwise loaded with fuel of the maximum permissible reactivity allowable under the burnup and enrichment curve. A maximum reactivity fresh fuel assembly for Harris would be a Westinghouse 15 × 15 PWR fuel assembly enriched to 5% (by weight) uranium-235. The analysis considered the presence of 2000 parts per million (ppm) of soluble boron in the

pool water, as required by Harris operating procedures. Furthermore, the analysis also evaluates criticality safety for two additional boron concentrations: (a) 400 ppm of soluble boron to confirm CP&L statements in its June 14, 1999 response to a Staff requests for additional information (RAI); and (b) 0 ppm of soluble boron. While not considered a credible scenario, CP&L states this analysis for zero boron concentration was performed to render moot any further discussion of the loss of soluble boron relative to this issue.

CP&L asserts that the results of this analysis demonstrate that a single fuel assembly misplacement, involving a fuel element of the wrong burnup or enrichment, will not cause criticality in Harris spent fuel pools C and D.[8] The analysis demonstrates that the spent fuel in the storage racks, with the required 2000 ppm of soluble boron in the SFP water, will remain subcritical at a *k*-effective of 0.7783 following the misplacement of a fresh fuel assembly with the maximum permissible enrichment at Harris. The analysis also demonstrates that the spent fuel in the storage racks will remain subcritical, with a *K*-effective of 0.9352, following a misplacement event assuming only 400 ppm of soluble boron is present in the SFP water. Finally, CP&L claims the analysis demonstrates that the spent fuel in the storage racks for Harris pools C and D will remain subcritical following a fresh fuel assembly misplacement event even if no soluble boron is present in the spent fuel pool water, with a *K*-effective of 0.9932.

CP&L states that these results affirmatively demonstrate that (1) CP&L has performed a criticality analysis of a single fuel assembly misplacement, involving a fresh fuel assembly with the maximum permissible reactivity at Harris, for the spent fuel storage racks in Harris pools C and D; and (2) the criticality analysis demonstrates that a single fuel assembly misplacement, involving a fresh fuel assembly fuel element with maximum permissible reactivity at Harris, will not cause criticality in Harris pools C and D. CP&L concludes that because the two questions it posed have been answered affirmatively, and BCOC does not dispute those answers, the Board should dispose of basis two of contention TC-2 in its favor.

Additionally, CP&L makes the following observations to bolster its argument that the likelihood of misplacement of a single fuel assembly is very small: (1) fresh fuel assemblies are first handled dry, in open air, and only then are positioned in pool A, which is located near Harris Unit 1 some distance from SPFs C and D; (2) due to financial considerations, there are usually only fifty-seven fresh assemblies on site at anytime; (3) proposed Harris technical specifications will prohibit loading of fresh fuel assemblies in pools C and D; and (4) information on fresh fuel movements is independently verified through two sources and is also tracked in a QA computer database.

CP&L also disputes BCOC's assertions that *K*-effective should be kept below 0.95 for all conditions according to applicable NRC and industry standards. CP&L asserts that 10 C.F.R. § 50.68(b)(4) should be the ultimate guidance on this subject and it permits a *K*-effective value above 0.95 (at a 95% probability, 95% confidence level), as long as it remains below 1.0, when credit is taken for soluble boron and when the spent fuel pool is accidentally flooded with pure water.

Finally, CP&L asserts that several new issues were raised by BCOC that should be dismissed. As has already been discussed, even though CP&L believes the first issue — the need for an evaluation of the loss of all soluble boron in the pool water concurrent with a fuel assembly misplacement — is not required under NRC regulations, it notes it has performed an analysis that demonstrates the spent fuel storage racks for Harris pools C and D will remain subcritical (*K*-effective of 0.9932) for this scenario. CP&L maintains this issue is resolved.

The same is true for the second issue — need to evaluate the concurrent misplacement of multiple fuel assemblies. While maintaining such a study is not required under NRC regulations, CP&L cites the results of a November 1999 Staff analysis demonstrating that, at a boron concentration of 2000 ppm, the spent fuel storage racks for Harris SFPs C and D will remain subcritical (*K*-effective of 0.98) when the storage racks are filled entirely with misloaded fresh fuel assemblies. Therefore CP&L maintains this issue is now resolved as well.

The third new issue — the need to analyze the universe of scenarios involving two or more unlikely, independent, or concurrent events — also is without substance according to CP&L. CP&L asserts that, as with the first two new issues, BCOC's requested analysis is not required under the DCP. Moreover, according to CP&L, in light of the criticality analyses CP&L and the Staff have already performed, BCOC has admitted that the only scenario missing from its "universe" of scenarios of two or more failures is multiple fuel assembly misplacement. Thus, BCOC's narrowing of the remaining universe of scenarios down to multiple fuel assembly misplacement renders the third new issue, as a practical matter, identical to the second issue, which the Staff's additional criticality analysis renders moot by demonstrating that the spent fuel storage racks for Harris SFPs C and D will remain subcritical following a misplacement that involves all fresh fuel assemblies.

c. *Staff Position*

The Staff's January 4, 2000 summary is supported by the affidavits of Dr. Kopp and NRC nuclear engineer Anthony P. Ulses. In general the Staff agrees with CP&L that there is no genuine and substantial factual dispute relating to basis two of contention TC-2.

[8] CP&L states that the methodology, assumptions, and results of the Harris Misplacement Analysis were reviewed and approved under the quality assurance requirements of both Holtec and CP&L. CP&L argues that these quality assurance reviews of the analysis by qualified nuclear criticality analysts provide reasonable assurance that the analysis results are valid. It also notes that Dr. Thompson, as BCOC's expert, did not challenge the validity of the analysis.

According to the Staff, it has reviewed the criticality calculations performed by CP&L, including the Harris Misplacement Analysis, and found them adequate. The Staff notes that Holtec International, which performed the CP&L analysis of reactivity effects for the proposed use of Harris pools C and D, analyzed reactivity effects of fuel storage in the Harris spent fuel racks using CASMO-3, which is a two-dimensional transport theory code. Holtec also used CASMO-3 for burnup calculations and for evaluating small reactivity increments associated with manufacturing tolerances. On the other hand, 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. Holtec also used MCNP-4A for independent verification calculations against CASMO-3.

According to the Staff, the CASMO-3 and MCNP-4A codes are widely used for analyzing fuel rack reactivity and have been benchmarked (i.e., compared to known values to evaluate their predictions) against results from numerous criticality experiments. The Staff declares that these individual analysis methods, which attempt to simulate the Harris spent fuel racks as realistically as possible for important parameters such as enrichment, assembly spacing, and absorber thickness, showed good agreement with each other. The Staff also maintains that comparison of different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. Moreover, these methods have been used and approved by the Staff in numerous other SFP criticality analyses.

Like CP&L, the Staff indicated it considers a fuel assembly misplacement unlikely, citing several reasons that generally agree with the CP&L arguments. First, the Staff notes that proposed Technical Specification 5.6.1.2 will control fuel storage limitations, and Harris selection procedure NFP-NGGC-0003 will be in place to control fuel assembly selection and avoid mispositioning errors. The Staff also observes that fresh fuel assemblies have a bright metallic color and are distinguishable from spent fuel assemblies, which have a darker, reddish color due to oxidation of the cladding, thereby providing a visual distinction that will help avoid misplacement errors. Third, the Staff notes that the proposed burnup limit curve is conservatively based on a minimum required burnup. Accordingly, unless a fuel assembly is prematurely discharged from the reactor, it will have a higher burnup than the burnup requirements and, therefore, a lower reactivity.

Also like CP&L, the Staff considers boron dilution events in pools C or D unlikely. Initially, the Staff argues that Harris Chemistry and Radiochemistry Procedure CRC-001 requires that boron concentration be kept at between 2000 and 2600 ppm, and that confirmation be done by monthly surveillance. Further, according to the Staff, Harris technical specification 3.9.11 requires a minimum of 23 feet of water above the top of the fuel rods, which provides adequate margins against water leakage or overfill. Additionally, in place to avoid boron dilution incidents are high and low water level alarms at the pools, as indicated by section

9.1.3 of the Harris final safety analysis report (FSAR). Finally, the Staff notes that a visual inspection of SFP water is done during each Harris operating shift.

Also significant, according to the Staff, is the November 1999 independent analysis it performed to assess the impact of misloading spent fuel pools C and D entirely with fresh fuel assemblies. For purposes of this analysis, the Staff assumed that soluble boron concentration was 2000 ppm, the pool water temperature was 4° Celsius, and there would be the worst conceivable misloading, consisting of Westinghouse 15 × 15 assemblies enriched to 5% U-235 without burnable poisons, which would be bounding as the highest allowed enrichment for commercial power reactor fuel. The Staff further states that it modeled the rack, fuel, and poison plate geometry using their nominal dimensions.

The Staff declares that it used the SCALE code system to perform the analysis, which it claims without dispute from BCOC has been validated for these types of calculations. According to the Staff, it further assumed that the storage racks were filled entirely with misloaded assemblies. The Staff asserts that such misloading, which could result only from multiple unlikely events requiring multiple errors, results in a predicted maximum *k*-effective of 0.98. The Staff concludes that because this configuration, which represents the worst possible series of misloading events, resulted in a *k*-effective of less than 1.0, the misloading of an entire rack of fresh fuel in spent fuel pools C or D will not lead to criticality.

d. Board Ruling[9]

The Board observed that basis two of contention TC-2 raised the following question of fact:

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

LBP-99-25, 50 NRC at 36. BCOC has suggested that, in making this statement, we misspoke relative to the DCP. In this regard, we note that as the basis for

[9] BCOC proffered Dr. Gordon Thompson as its expert witness for this contention. Citing various agency precedent regarding the qualifications of expert witnesses, the Staff maintains that Dr. Thompson does not qualify as an expert witness by virtue of his knowledge, skill, experience, training, or education. According to the Staff, because Dr. Thompson is no more qualified to render an expert technical opinion on criticality than any layperson, any conclusions he makes, opinions he renders, or other testimony related to this contention should be stricken. *See* Staff Summary at 14-19.

After hearing party presentations regarding this objection during the January 21, 2000 oral argument, *see* Tr. at 207-18, the Board ruled from the bench that it would not declare Dr. Thompson ineligible to be the BCOC expert on this matter, but would assign his testimony appropriate weight commensurate with his expertise and qualifications, *id.* at 441. In this regard, we note that by reason of his experience and training, his expertise relative to reactor technical issues seems largely policy-oriented rather than operational.

its contention, after quoting the DPC provision of draft Regulatory Guide 1.13, BCOC stated:

> CP&L's proposed administrative controls on criticality would not satisfy this requirement because only one failure or violation, namely placement in the racks of PWR fuel not within the "acceptable range" of burnup, could cause criticality.

[BCOC] Supplemental Petition to Intervene (Apr. 5, 1999) at 13. Relative to BCOC's concern, although we believe our statement is a fair characterization of its position at that time, we will not, as CP&L and the Staff suggest, reject any consideration of a multiple fuel misplacement scenario.

Be that as it may, the Board finds that the analyses performed by CP&L and the Staff have adequately answered the question posed by this portion of the contention, namely, would fuel assembly misplacement, involving fuel assemblies with the maximum permissible enrichment, cause SFP criticality. Specifically, the CP&L SFPs C and D criticality calculations involving the misplacement of a fresh fuel assembly with the maximum permissible reactivity, the technical details and computational accuracy of which BCOC has not contested, demonstrate that with respective *K*-effectives of 0.7783 and 0.9352, the pools would not go critical when the boron concentration in the water is at the required minimum level of 2,000 ppm or at a significantly lower level of 400 ppm. Moreover, as the study demonstrates, this is true even if there were no boron in the spent fuel pools, which produces a *K*-effective of 0.9932. This clearly provides an upper bound for the criticality analyses of misplacement of a single fuel assembly concurrent with an accidental loss of some or all of the SFP's soluble boron.

The Staff also performed a further independent analysis that shows that, with boron at the minimum required level, even misplacing all fuel assemblies in the pool would cause a *K*-effective of 0.98, which would not cause spent fuel pools C or D to go critical. Again, BCOC has not disputed the technical details and computational adequacy of the Staff's calculations for this postulated scenario.

BCOC did advance a theory in its oral argument that *K*-effective must be kept at or below 0.95 under all conditions, including the scenario in which a fresh fuel assembly is misplaced concurrent with an accidental loss of all soluble boron. Such a theory is meritless, however, in the face of 10 C.F.R. § 50.68(b)(4), which states in pertinent part:

> If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The intent of this requirement is unambiguous. *K*-effective must be kept at 0.95 or below when credit for soluble boron is taken; if, however, there is the accidental loss of boron, the SFP still cannot go critical, i.e., it must remain below a *K*-effective of 1.0. Thus, there is no requirement that *K*-effective must be kept at or below 0.95 under all conditions, including the scenario involving a fresh fuel assembly misplacement concurrent with the loss of soluble boron.

Additionally, though they are not central to the resolution of basis two, the Board also finds credible (a) the evaluation proffered by CP&L and the Staff indicating a low likelihood of a fresh fuel assembly misplacement in SFPs C and D; and (b) the evaluation provided by CP&L and the Staff indicating a small probability that boron dilution will occur in spent fuel pools C or D. Supporting our conclusion relative to item (a), above, is the combination of (1) measures involving technical specifications requirements and procedural controls; (2) the use of independent verification for fuel movement; and (3) the visual differentiation of spent fuel and fresh fuel assemblies, all of which lead to a low likelihood of misplacing a fresh fuel assembly. And for item (b) above, based on (1) the technical specification requirements and procedural controls regarding SFP boron concentration, (2) the margins inherent in the 23 feet of water above the fuel assemblies, (3) the existence of high and low water level alarms, and (4) the visual checks during each shift of operation, the Board similarly is satisfied that the probability of a boron dilution event is small.

Finally, relative to the "new" issues raised by BCOC during discovery as delineated by CP&L in its January 4, 2000 filing and addressed in detail by both CP&L and BCOC during the January 21, 2000 oral argument, involving (a) the loss of all soluble boron concurrent with the misplacement of a fuel assembly, (b) concurrent misplacement of multiple fuel assemblies, and (c) the analysis of scenarios of two or more unlikely, independent, concurrent events, we find that each has been adequately resolved or rendered moot by the analyses performed by CP&L and the Staff.

As a consequence, in accordance with 10 C.F.R. § 2.1115(a), we find relative to this portion of contention TC-2 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing. As such, based on the record before us, we dispose of this portion of the contention as being resolved in favor of CP&L.

B. Contention TC-3

The substance of Contention TC-3 consists of disputes over five matters:

1. What equipment within the spent fuel pool cooling and cleanup system (SFPCCS) and the component cooling water system (CCWS) is covered relative to BCOC's quality assurance concerns?

2. Whether the proposed activation of equipment complies with the requirements of 10 C.F.R. Part 50, Appendix B?

3. Whether the CP&L proposed Alternative Plan is adequate to meet the requirements of 10 C.F.R. § 50.55a(a)(3)?

4. What are the consequences of a failure of the equipment covered?

5. Does the nature of the proposed change to the facility require that a construction permit be issued?

We treat each issue below.

1. Scope of the Equipment Covered by the Contention

DISCUSSION: CP&L Summary at 76-77; Staff Summary at 49-50; Tr. at 325-26, 346-47, 382.

a. Parties' Positions

In its January 4, 2000 written statement, referencing the deposition of Union of Concerned Scientists nuclear safety engineer and BCOC supporting expert David Lochbaum, the Staff noted Mr. Lochbaum's agreement that the only equipment issues in contention were the fifteen welds in the piping embedded in concrete so as not to be subject to inspection from the outside. CP&L likewise noted that Mr. Lochbaum had "conceded that the SFPCCS heat exchangers, pumps, and accessible piping . . . are not at issue in Contention [TC-]3." CP&L Summary at 76. BCOC had not addressed this matter directly in its written statement; however, at the January 21, 2000 oral argument BCOC asserted that "the scope of the equipment that has not been kept in an appropriate lay-up condition at Harris over the last 15 or so years is broader than the scope of equipment as defined in [BCOC]'s contention." Tr. at 325. BCOC also argued "that, in fact, other equipment was not kept in an appropriately laid-up condition" and asserted that "we are planning to file a request for an amendment of the contention to seek restoration of that part of the contention that was dropped." *Id.* at 326.

b. Board Ruling[10]

BCOC has not filed a request to amend its contention to seek to further define the scope of equipment covered. Accordingly, in light of the statements of BCOC

[10] As with Dr. Thompson, *see supra* note 9, the Staff initially sought to disqualify Mr. Lochbaum as an expert witness and have his testimony relative to contention TC-3 stricken or limited. *See* Staff Summary at 65-66. At the oral argument, however, the Staff amended its motion to request that the Board assign his testimony appropriate

(Continued)

witness Mr. Lochbaum, the Board limits its consideration of this contention to the condition of the fifteen welds and associated piping that are inaccessible because they are embedded in concrete.

Thus, in accordance with 10 C.F.R. § 2.1115(a), we find relative to this portion of contention TC-3 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

2. Compliance with 10 C.F.R. Part 50, Appendix B.

DISCUSSION: Detailed Summary of Facts, Data and Arguments and Sworn Submission on Which [BCOC] Intends to Reply at Oral Argument To Demonstrate the Existence of a Genuine and Substantial Dispute of Fact with [CP&L] Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with Respect to Quality Assurance Issues (Contention TC-3) (Jan. 4, 2000) at 16-24 [hereinafter BCOC TC-3 Summary]; Staff Summary at 51-53; Tr. at 330-32, 356-58, 396-97.

a. Parties' Positions

As we noted in Section I.B above, this contention challenges CP&L's compliance with the requirements of 10 C.F.R. Part 50, App. B, the Commission's quality assurance regulation, and in particular its adherence to Criteria XIII, XVI, and XVII governing, respectively, storage and preservation of equipment, measures to correct damage or deterioration, and record keeping. Indeed, in admitting the contention the Board stated:

> It is also clear from the positions of all the participants that some of the piping and equipment have not been properly stored and proper records regarding its quality during that period have not been maintained. Whether such storage and maintenance are necessary as a matter of law and fact is clearly a subject of dispute among the participants. The argument concerning this point is not a simple one . . . .

LBP-99-2, 50 NRC at 37.

The Staff argues that the requirements of Appendix B only apply during construction and operation and that, in effect, since the Harris construction permit expired and the SFPCCS for pools C and D was never part of an operating plant

weight commensurate with his expertise and qualifications, Tr. at 393-95. We do so here, noting that in this context his qualifications appear to run to facility procedures and operations, e.g., whether a particular procedure to detect microbiologically induced corrosion (MIC) was properly utilized, rather than substantive knowledge of the underlying technical subject involved with the procedure, e.g., whether a claimed piping defect was MIC. *See* Tr. at 334.

"CP&L does not have to demonstrate compliance with Appendix B during the lay-up period." Staff Summary at 52. CP&L takes the same position, namely that at the time Harris Unit 2 construction was abandoned, the piping and welds were no longer under construction, were not in operation, and had no safety-related function. As a consequence, CP&L maintains, by its own terms Appendix B did not apply during the post-abandonment period. CP&L and the Staff thus would have us find that the lack of compliance with Appendix B during the layup period is of no consequence. Instead, in their view, all that matters is whether CP&L's Alternative Plan, submitted under 10 C.F.R. § 50.55a(a)(3) as offering an alternative to the code requirements therein, is sufficient to provide an acceptable level of quality and safety.

In contrast, BCOC maintains that CP&L's preparation of an alternative plan to conform to the requirements of section 50.55a simply goes to the question of the pedigree of the piping, i.e., to compensate for the fact that the original quality assurance documentation has been lost in a number of instances. It does not, however, excuse CP&L from a showing of compliance with the terms of Appendix B for that piping during the period of abandonment.

b. Board Ruling

In the Board's view, in the context of this amendment request, the evident purpose of both regulatory provisions is so closely parallel that we can regard compliance with section 50.55a as affording compliance with Appendix B. If the CP&L Alternative Program complies with section 50.55a, it is acceptable under Appendix B as well.

We thus will proceed to analyze the extent to which the CP&L Alternative Plan represents a proper alternative under the requirements of section 50.55a, confident that if its coverage is appropriate, compliance with Appendix B will have been achieved.[11]

Pursuant to 10 C.F.R. § 2.1115(b), we find relative to this portion of contention TC-3 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

3. Adequacy of the Alternative Plan To Meet the Requirements of 10 C.F.R. § 50.55a(a)(3)

DISCUSSION: BCOC TC-3 Summary at 24-51; CP&L Summary at 78-89; Staff Summary at 55-65; Tr. at 339-43, 360-75, 397-405.

a. BCOC Position

BCOC's attack on the CP&L amendment request under this portion of contention TC-3 centers on both the notion that CP&L has only a "snapshot" of the conditions under which storage took place and what BCOC describes as the Staff's "Sleeping Beauty" notion, i.e., that the "CP&L went to sleep for 15 years" and that, once it awakened from its slumber, it was as if all those years had never passed relative to the piping systems at issue. Tr. at 329. With Mr. Lochbaum as its supporting witness, BCOC challenges the crux of the CP&L and Staff reasoning that a combination of construction period QA (to the extent that QA can now be verified) and present-day inspection and testing of the interior of the piping at issue can serve as a substitute for a QA program carried out with continuity throughout the construction and operation of the equipment at bar. In this regard, BCOC questions whether the construction period QA can be proven adequate absent certain construction era documentation and whether the current inspection of the embedded welds is complete and sufficient.

b. CP&L Position

CP&L observes that "[t]he 50.55a Alternative Plan addresses the existing situation where [the Harris facility] is no longer under construction, CP&L no longer maintains its ASME N-Stamp certification program, and certain quality documentation was discarded concerning field welds." CP&L Summary at 78. Further, CP&L asserts that "BCOC has not challenged the adequacy of the Supplemental QA requirements as an alternative to ASME N-Stamp certification." *Id.* at 79.[12]

[11] Relative to the specifics of compliance with 10 C.F.R. Part 50, App. B, BCOC argues that CP&L has failed to meet Criterion XIII, requiring measures to control handling, shipping, and storage. *See* BCOC TC-3 Summary at 16-19. We note, however, that a substantial portion of the Alternative Plan is devoted to showing that deterioration did not occur during the layup period as a substitute for this requirement, and BCOC presents no expert testimony showing why the Alternative Plan has not demonstrated adequate evidence of this equivalence. Similarly, BCOC alleges that appropriate corrective actions have not been taken in accord with Criterion XVI of Appendix B, but presents no expert testimony as to what corrective actions are necessary. *See id.* at 20-23. BCOC further states that the Alternative Plan does not describe what criteria are to be used in inspecting piping and welds, and that the reader of the Alternative Plan "reasonably presumes that the criteria must relate to the piping pedigree, not to its condition." *Id.* at 22. In fact, the BCOC summary subsequently cites such criteria. *See id.* at 41.

[12] In a footnote to that statement, CP&L asserts that the only facts that BCOC presents attacking the Harris facility QA program for the construction period are four NRC inspection reports from 1981, which are mentioned by Mr. Lochbaum in his affidavit. CP&L points out, however, that in his deposition, Mr. Lochbaum asserted that the minor infractions that these reports addressed "wouldn't lead me to believe that the quality assurance program at Shearon Harris was deficient or had a programmatic breakdown." CP&L Summary at 79 n.202 (quoting *id.* exh. 10, at 129-30 (Lochbaum Deposition (Oct. 24, 1999))).

Accordingly, at the outset, CP&L's position is that "[t]he acceptability of the embedded welds in 1983 . . . has been demonstrated by the implementation of the 'Piping Pedigree Plan.'" *Id.* at 81. To buttress this position, CP&L offers the affidavits of CP&L spent fuel project manager Edwards, and of CP&L employees David L. Shockley, Charles H. Griffith, and William T. Gilbert. The latter three affiants, who worked at the Harris plant during the construction period and attest to familiarity with quality assurance matters relating to the embedded piping and associated welding at issue, assert that the procedures in effect during construction made certain that the fifteen welds in question were, in fact, completed in accordance with the QA program then in effect. They base their conclusions on procedures in effect at the time that required certain inspections to be completed before concrete pours and hydrostatic tests, signatures of authorized nuclear inspectors, and on the presence of their own signatures

ASME Code requirements sufficient to demonstrate that the welds and piping are acceptable for service. Referencing the affidavits of Mr. Heck and Mr. Naujock, the Staff outlines the results of its review process regarding existing construction records. Mr. Heck asserts that the sequential QA requirements and signatures of QA personnel at various hold points in the process, such as hydrostatic testing and concrete placement, give confidence that all welds were done in compliance with ASME and other QA requirements, concluding "the subject welds were completed with an acceptable level of quality and safety." Staff Summary, Affidavit of Kenneth C. Heck in Support of NRC Staff's Written Summary (Jan. 10, 2000) at 27. The Staff's affiant Naujock also concludes that the welds made on SFPs C and D piping "were made by qualified personnel using qualified procedures in accordance with the objectives of [ASME Code] Section III requirements." *Id.* Affidavit of Donald G. Naujock in Support of Brief and Summary of Relevant

construction, nor does it seem that he has any independent knowledge of such flaws.

Thus, on the matter of the adequacy of the original construction, the record before us fully supports the conclusion that the piping and associated welds at issue were completed in accordance with the agency QA regulations and applicable ASME code requirements.

This brings us to the question embodied in BCOC's "snapshot" and "Sleeping Beauty" allegations: Will the assurance that the original QA program was adequate, when coupled with the present-day procedures and tests embodied in the Alternative Plan, give assurance that the present piping/weld quality is adequate, despite the long period when the equipment was not subject to storage and inspection conditions that were strictly in accordance with QA procedures.

We have accepted CP&L's positions that the fuel pool piping was built to agency QA and ASME Code requirements and that the contention at bar relates only to the fifteen welds and associated piping that are inaccessible because they are embedded in concrete. Bearing in mind these findings, to answer the question posed above we must review the adequacy of the current tests and procedures relating to the embedded material. In this regard, CP&L points out that "[t]he tests and inspections included testing of the water in the SFPCCS piping, a complete walk-down and visual inspection of all accessible piping, welds, components and equipment, re-inspection of all accessible welds, testing the weld filler material in the accessible welds" and inspection of the surface of the spent fuel pool walls and concrete in which the piping is embedded to detect any evidence of outside chemical attack on the external surface of the piping. CP&L Summary at 93.

Looking to these activities, several points stand out. Regarding the water that has been in the SFPCCS during the layup period, it was analyzed by Harris chemists for chemical content and by Dr. Ahmad Moccari for microbiological content. The water turned out to be of high purity and did not contain any bacteria capable of causing microbiologically induced corrosion (MIC). The results of this testing indicated a highly unlikely potential for chemically or microbiologically induced corrosion according to CP&L's expert on corrosion, Dr. Ahmad Moccari. *See* CP&L Summary at 93-94.

Additionally, all fifteen embedded welds and their associated piping were inspected using a high resolution camera, taking high quality pictures of everything inside the piping, longitudinal welds, circumferential welds, and piping surfaces. *See id.* at 94-95. Some general discoloration of welds and piping was noted. Reddish brown deposits were observed on welds and piping, as were shallow indications on a weld and seam, and incomplete melting of some consumable inserts. *See id.* at 95-96. Samples of the reddish brown deposits were taken and the remaining deposits were removed with high pressure water and the surface reinspected. While Mr. Licina noted what appeared to be two small pits under the deposit, both he and Dr. Moccari agreed that these pits would have no impact on the integrity of the piping. *See id.* at 96. The deposit material was analyzed with a scanning electronic microscope and found to consist of iron oxide, similar in appearance to that introduced into the spent fuel pool water during the transshipment of fuel from other CP&L plants. The material, however, neither results from, contributes to, nor is otherwise associated with corrosion or degradation of the piping. *See id.* at 97. The deposits simply represent places at which crud accumulated. *See id.* at 98. The incomplete melting of the consumable inserts was not viewed by CP&L's experts as cause for concern. *See id.* at 99-100.

The largest of the shallow indications mentioned above was about ½ inch long. Since the chemical and temperature conditions are not aggressive and the line was not exposed to thermal or loading cycling, the specific cause of this indication could not be determined. *See id.* at 101-02. However, CP&L's contractor SIA independently evaluated the implication of such indications for the structural integrity of the piping and concluded that they did not pose any challenge to that integrity or to the piping's suitability for service. *See id.* at 102.

The Staff has also evaluated CP&L's Alternative Plan, the analyses and examinations it calls for, and the environment and present condition of the embedded pipes and welds. *See* Staff Summary at 63. The Staff's evaluation was conducted by Mr. Naujock and Dr. Davis, whose affidavits are proffered as a foundation for the conclusion that "the welds and piping are acceptable for service and that the Alternative Plan provides an[] acceptable level of quality and safety." *Id.* at 65. A significant element in this conclusion, it seems apparent, was the CP&L summer 1999 visual inspection of the interior surface of the embedded welds using a high resolution remote camera capable of detecting a 1-mil diameter wire and demonstrated to be capable of detecting small flaws consistent with ASME Code requirements. The Staff notes that enhanced visual inspection has been approved in previous cases for reactor vessel internals. *See id.* at 63-64.

Staff expert Dr. Davis reviewed the videotapes resulting from the remote camera examinations of ten of the fifteen embedded welds and observed no evidence of MIC, no degradation of the welds, and nothing that required corrective action. This Staff expert noted that five of the welds required further evaluation and that these welds were analyzed by SIA. From the review and analysis of the videotapes and from the available documentation, the Staff concluded that the piping and welds are conservatively designed; are several times thicker than required by ASME Code; are generally in good condition with some minor, but no major defects; and have leaktight integrity. The Staff also concluded that there were no viable mechanisms for longitudinal cracking such as intergranular stress corrosion cracking, transgranular stress corrosion cracking, or localized corrosion. The only mechanism the Staff could find viable for corrosion was MIC, and the water sampling and sampling of deposits on one weld produced no evidence of that. Further, no leaks consistent with MIC were observed on any of the accessible piping. The Staff thus determined that the welds were completed with an acceptable

level of quality and safety and that no degradation of the welds and pipes occurred during the layup period. *See id.* at 64-65.

For its part, BCOC would have us find that the videotapes revealed evidence of degradation and that evidence was not adequately investigated. *See* BCOC TC-3 Summary at 4. In this regard, BCOC relies on a deposition statement by its expert witness, Mr. Lochbaum, concerning certain details from the remote camera videotapes for the proposition that there were "shop welds" present in addition to the "field welds" and that fact represents some sort of deficiency in the entire procedure. *See id.* at 44-45. BCOC argues that a shop weld was discovered by accident, and that only that one weld, and none of the many other shop welds, was examined. *Id.* At oral argument, CP&L pointed out that the remark cited by Mr. Lochbaum was, later in the videotape, found to pertain to a field weld, and that the camera operator, who was not an expert in interpreting the results, was not criticizing the condition of the weld but was simply making an irrelevant remark because of a mistake in the position of the camera. Further, Staff witness Dr. Davis and CP&L witness Mr. Edwards examined the weld in question and found no fault with it. *See* Tr. at 368-69. And, in addressing this point, CP&L also declared that the pedigree of every shop weld was available (because, of course, that pedigree had been established at the fabricator's plant and was not part of the documentation discarded after suspension of construction). And shop welds are, in any event, less susceptible to corrosion than field welds. *See* Tr. at 432-34.

In sum, CP&L and the Staff again have the better of the argument. Those with expertise in the fields of corrosion, welding, and ASME Code requirements attest on behalf of these two parties that the procedures that were used to substitute for construction records and examination during layup are adequate to assure a level of safety as required by the regulations. *See* Tr. at 404-05. Moreover, even BCOC's witness Mr. Lochbaum, when asked what he would require in the Alternative Plan to satisfy his concerns replied, "[a] complete visual inspection of the interior piping surfaces, all of the welds of the embedded portions, and some evaluation, analysis or inspection of the exterior piping surfaces." CP&L Summary, exh. 10, at 218-19 (Lochbaum Deposition). The record established that is just what has been done to document the Alternative Plan's compliance with section 50.55a.

We find, therefore, that the Alternative Plan is adequate to satisfy the applicable requirements of the regulations including those of 10 C.F.R. § 50.55a(a)(3), and 10 C.F.R. Part 50, App. B. Further, in accordance with 10 C.F.R. § 2.1115(a), we find relative to this portion of contention TC-3 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

4. *Consequences of Piping or Weld Failure*

DISCUSSION: CP&L Summary at 104; Staff Summary at 66-69; Tr. at 375-8 411-14, 431.

a. *Parties' Positions*

In its written summary, BCOC did not address the question of the safe implications of piping or weld leakage. Nor did BCOC's technical witness, M Lochbaum, postulate any complete scenario that would lead to loss of coolin damage to safety equipment, or releases to the environment during his Octob 1999 deposition or otherwise. Indeed, when questioned about the possib consequences of leakage, Mr. Lochbaum could not point to any precise scenar in which leakage could be large enough to interfere seriously with system functio or release contaminants to the environment. At oral argument, however, BCO briefly addressed the matter, arguing that tritium leakage has occurred from spe fuel pools at other facilities and that all equipment should conform to quali requirements.

CP&L asserts that since the piping is embedded in reinforced concrete, the is no way for a leak to result in loss of water that even approaches the norm evaporation rate from the pool, that there is an entirely redundant run of pipin to carry water in the event of a broken pipe, and that there is no pathway the environment. When, at oral argument, the Board pursued the question possible leakage into sensitive equipment, CP&L explained there was no equipme that pool water could leak into that would compromise the safety of the plan Further, during oral argument CP&L showed diagrams of the fuel pool buildin that indicated that leakage would not have any path to the environment, b would be captured by floor drains and diverted to the plant's waste processin system. And under Board questioning concerning a historical incident in whic pool water contaminated the environment by leakage, CP&L explained how th instant circumstances were substantially different in matters involving pool an building design from those mentioned by the Board.

For its part, the Staff offered the affidavit of NRC reactor systems engine Christopher Gratton addressing the question of whether the failure of the wel or piping could result in a hazard affecting public health and safety. The Sta argues that such a result is unlikely and concludes whether or not leakage is abl to flow out of the pool's concrete structure, a break in the embedded piping welds whose leakage is within the coolant systems makeup capacity would ha a minimal effect on the operation of the system, the coolant inventory, or th safety of the stored fuel. *See* Staff Summary at 67. Moreover, according to th Staff, even if substantial leakage were to occur, the position of the pools' pipin penetrations is such that only forced cooling would be lost, the pool level woul

remain well above the stored fuel, and the rate of boil-off would be well within the capacity of available coolant makeup systems. The Staff thus concludes that the stored fuel would remain covered and cooled with only a minimal impact on safety.

b. Board Ruling

In theory, the leakage from a spent fuel pool could be so severe that the cooling system would become inoperable, either from low water level or ruptured pipes; the leakage could result in the release of pool water (presumably contaminated at least with tritium) to the environment; or the leakage could penetrate safety-related equipment and cause it to malfunction. Based on the record now before us, however, it is clear that the result of any weld or piping failure at Harris could have only a limited number of effects on the integrity of the plant and the health and safety of the public.

The Board has already found that the CP&L Alternative Plan offers quality assurance and safety equivalent to the requirements of the regulations. And for its part, BCOC has offered no reason to suggest that a leak from the welds or piping at issue would in any way parallel the SFP leakage situations it relied upon at the January 21, 2000 oral argument. Indeed, upon Board inquiry regarding one of the pools mentioned by BCOC, CP&L described in some detail the reasons why that leakage situation differed from circumstances at the Harris plant.

To be sure, BCOC accuses CP&L and Staff of ignoring the health significance of continuous small leaks in nuclear power plant piping. Yet, from the record before us, we have no reason to believe that small amounts of leakage, such as those which could occur from pinholes or hairline cracks in pipes embedded in concrete, would lead to any hazard to the plant or the public.

Consequently, in accordance with 10 C.F.R. § 2.1115(a), we find relative to this portion of contention TC-3 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

5. Need for a Construction Permit

DISCUSSION: BCOC TC-3 Summary at 16; Tr. at 200-01, 327-32, 348-53, 391-93, 417-18.

a. Parties' Positions

Although BCOC first raised this issue of whether a construction permit is needed to prepare and activate the spent fuel pool C and D systems in its written statement, it denies the question is a "new contention," asserting it was clearly embodied in contention TC-3 as admitted. Tr. at 327. According to BCOC, the need for a construction permit is connected to its assertion that the Applicant had not complied with 10 C.F.R. Part 50, App. B.

CP&L, however, claims that this issue is a new contention that has not been shown to meet the late-filing standards of 10 C.F.R. § 2.714(a)(1). CP&L also argues that it is not seeking either a construction permit or conversion of such a permit into an operating license, but rather a change in its operating license pursuant to 10 C.F.R. §§ 50.59, 50.90. Further, to CP&L's knowledge, no construction permit has been required for any of the large post-Three Mile Island accident changes or, indeed, for replacement of steam generators or power upgrades, and that there has never been a construction permit required for a change in an operating license applying to a commercial operating plant. In this regard, CP&L relies upon 10 C.F.R. § 50.92(a) that states "[i]f the application involves the material alteration of a licensed facility, a construction permit will be issued before the issuance of the amendment to the license."

CP&L further argues that this "material alteration" test has been interpreted as a change in the type of major components at an existing facility, a change that would introduce significant new issues relating to the function and nature of the facility and to the public health and safety. CP&L cites as precedent a Director's Decision, *Virginia Electric and Power Co.* (Surry Power Station, Units 1 and 2), DD-79-19, 10 NRC 625, 654-61 (1979), in which the Director of Nuclear Reactor Regulation found that the replacement of reactor steam generator internals did not rise to the significance of a "material alteration" to the plant. CP&L, in reply to a Board question, went so far as to say that even if the additional spent fuel pools had to be constructed "from scratch," that would not constitute a material alteration under the regulations. *See* Tr. at 351-52.

The Staff also regards the question of the requirement for a construction permit as embodying a new contention and as a matter that did not come out in discovery. The Staff agreed with CP&L that the amendment at issue does not represent a material alteration of the facility, noting that the Staff's review of the case law generally is in accord with CP&L's. Although the Staff equivocated somewhat on whether building new pools would require a construction permit, it agreed that steam generator replacement or the construction of new buildings does not require a construction permit, but rather could be accomplished by an amendment under 10 C.F.R. § 50.59.

b. Board Ruling

BCOC's claim that a construction permit is required for the CP&L request was not a part of the admitted contention. As such, it can only be admitted if it fulfills the section 2.714(a) late-filing standards, which BCOC has made no effort to

address. This precludes further consideration of the issue. *See Boston Edison Co.* (Pilgrim Nuclear Power Station), ALAB-816, 22 NRC 461, 465-68 (1985). Even if this claim had been within the scope of contention TC-3, however, under the circumstances here, we are skeptical that the amendment before us is a "material alteration" in the sense intended by the regulations so as to require a construction permit.

Once again, pursuant to 10 C.F.R. § 2.1115(b), we find relative to this portion of contention TC-3 that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, dispose of this portion of the contention as being resolved in favor of CP&L.

III. CONCLUSION

With respect to contention TC-2, Inadequate Criticality Prevention, the Board concludes that (1) Applicant CP&L's request to utilize credit for burnup and enrichment as criticality control measures is consistent with the requirements of 10 C.F.R. Part 50, App. A, GDC 62; and (2) the use of credit for burnup and enrichment does not violate the double contingency principle of draft Regulatory Guide 1.13. In connection with contention TC-3, Inadequate Quality Assurance, we conclude relative to the embedded welds and piping at issue, the CP&L Alternative Plan is sufficient under 10 C.F.R. § 50.55a(a)(3) and 10 C.F.R. Part 50, App. B, to provide an acceptable level of quality and safety. Further, as to both contentions, based on the record before us, pursuant to 10 C.F.R. § 2.1115(a), the Board further concluded that there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an evidentiary hearing and, based on the record before us, we dispose of those contentions as resolved in favor of CP&L.[14]

For the foregoing reasons, it is, this fifth day of May 2000, ORDERED that with respect to BCOC contentions TC-2, Inadequate Criticality Control, and TC-3, Inadequate Quality Assurance, in accordance with 10 C.F.R. § 2.1115(a), the Board concludes (1) there is no genuine and substantial dispute of fact or law that can only be resolved with sufficient accuracy by the introduction of evidence in an

[14] Although this ruling completes action regarding all the technical contentions before us relative to the December 1998 CP&L amendment request, because the admissibility of four BCOC late-filed environmental contentions is yet to be resolved, this proceeding is not subject to dismissal in accordance with 10 C.F.R. § 2.1115(a)(2).

evidentiary hearing; and (2) contentions TC-2 and TC-3 are disposed of as being *resolved* in favor of CP&L.

THE ATOMIC SAFETY AND
LICENSING BOARD[15]

G. Paul Bollwerk, III
ADMINISTRATIVE JUDGE

Frederick J. Shon
ADMINISTRATIVE JUDGE

Peter S. Lam
ADMINISTRATIVE JUDGE

Rockville, Maryland
May 5, 2000

[15] Copies of this Memorandum and Order were sent this date by Internet e-mail transmission to counsel for (1) Applicant CP&L, (2) Intervenor BCOC, and (3) the Staff.

applicable to nuclear power facilities.[1] *See* 10 C.F.R. Part 72. In its September 29, 2000 final Safety Evaluation Report, the NRC Staff found acceptable the alternative seismic analysis proposed by PFS.

The State of Utah contests the exemption request. It seeks to amend Contention Utah L (geotechnical) to encompass the issues raised by PFS's exemption request. The exemption and admissibility questions arise in that context.

The parties should submit briefs as follows:

All parties should submit electronic copies of briefs, with paper copies to follow, by March 2, 2001. The briefs should be no more than twenty pages in length. The briefs should address both the exemption and the admissibility questions.

Reply briefs should be submitted no later than March 12, 2001. Reply briefs should not exceed ten pages in length.

IT IS SO ORDERED.

For the Commission[2]

ANNETTE L. VIETTI-COOK
Secretary of the Commission

Dated at Rockville, Maryland,
this 14th day of February 2001.

[1] An ISFSI located west of the Rocky Mountain Front is required to meet the standards applicable to nuclear power facilities found in 10 C.F.R. Part 100, Appendix A. Appendix A calls for the use of a deterministic seismic hazard analysis. In 1997, Part 100 was amended to allow nuclear power reactor licensees to use a probabilistic analysis. *See* 10 C.F.R. § 100.23. The NRC Staff is currently revising Part 72 to conform to this change and allow new ISFSI licensees the option to use a probabilistic analysis. *See* "Rulemaking Plan: Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations, 10 C.F.R. Part 72," SECY-98-126. The exemption approved by the Staff, however, allows PFS to design the ISFSI to standards that do not conform to either Appendix A or to the proposed rule.

[2] Commissioner Diaz was not present at the affirmation of this Order. Had he been present, he would have affirmed his prior vote to approve this Order.

Cite as 53 NRC 113 (2001) CLI-01-7

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS:

Richard A. Meserve, Chairman
Greta Joy Dicus
Nils J. Diaz
Edward McGaffigan, Jr.
Jeffrey S. Merrifield

In the Matter of **Docket No. 50-400-LA**

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant)

February 14, 2001

The Commission denies the Intervenor's petition for review of the NRC Staff's no significant hazards consideration determination and issuance of a license amendment for spent fuel pool expansion prior to completion of a related 10 C.F.R. Part 2, Subpart K proceeding. The Commission, however, directs the Staff to provide additional information to enable the Commission to decide whether to take discretionary review of the Staff's action, and directs the Licensee temporarily not to store spent fuel under the license amendment.

AEA: LICENSING DECISION (IMMEDIATE EFFECTIVENESS); RIGHT TO HEARING

COMMISSION PROCEEDINGS: AMENDMENTS (NO SIGNIFICANT HAZARDS CONSIDERATION)

OPERATING LICENSE AMENDMENTS: NO SIGNIFICANT HAZARDS CONSIDERATION

RULES OF PRACTICE: OPERATING LICENSE AMENDMENTS

OPERATING LICENSES: AMENDMENTS

Under the so-called "Sholly Amendment," the Commission is authorized to issue immediately effective reactor license amendments, "in advance of the holding and completion of any required hearing," upon a "no significant hazards consideration" determination. *See* Atomic Energy Act § 189a(2)(A), 42 U.S.C. § 2239(a)(2)(A).

COMMISSION PROCEEDINGS: AMENDMENTS (NO SIGNIFICANT HAZARDS CONSIDERATION

OPERATING LICENSE AMENDMENTS: NO SIGNIFICANT HAZARDS CONSIDERATION

OPERATING LICENSES: AMENDMENTS

RULES OF PRACTICE: OPERATING LICENSE AMENDMENTS

The Staff is authorized to find that a license amendment involves no significant hazards consideration

> if operation of the facility in accordance with the proposed amendment would not:
> (1) [i]nvolve a significant increase in the probability or consequences of an accident previously evaluated; or
> (2) [c]reate the possibility of a new or different kind of accident from any accident previously evaluated; or
> (3) [i]nvolve a significant reduction in a margin of safety.

See 10 C.F.R. § 50.92(c).

RULES OF PRACTICE: APPELLATE REVIEW

Our regulations provide that "[n]o petition or other request for review of or hearing on the staff's significant hazards consideration determination will be entertained by the Commission." *See* 10 C.F.R. § 50.58(b)(6).

OPERATING LICENSE AMENDMENTS: NO SIGNIFICANT HAZARDS CONSIDERATION

RULES OF PRACTICE: APPELLATE REVIEW; FINALITY OF DECISIONS

COMMISSION AUTHORITY OVER STAFF ACTIONS

NRC: AUTHORITY

RULES OF PRACTICE: *SUA SPONTE* REVIEW

The NRC Staff's determination on the no significant hazards consideration issue is final, "subject only to the Commission's discretion, on its own initiative, to review the determination." *See* 10 C.F.R. § 50.58(b)(6).

COMMISSION AUTHORITY OVER STAFF ACTION

NRC: SUPERVISORY AUTHORITY

RULES OF PRACTICE: *SUA SPONTE* REVIEW

The Commission has inherent authority to exercise its discretionary supervisory authority to stay the NRC Staff's actions or rescind a license amendment. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-86-12, 24 NRC 1, 4-5 (1986), *rev'd and remanded on other grounds, San Luis Obispo Mothers for Peace v. NRC,* 799 F.2d 1268 (9th Cir. 1986).

OPERATING LICENSE AMENDMENTS: NO SIGNIFICANT HAZARDS CONSIDERATION; DISPOSAL OF SPENT FUEL

In enacting the "Sholly Amendment" in 1983, Congress expressed "special concerns about significant hazards considerations for spent fuel license amendments." *See Diablo Canyon,* CLI-86-12, 24 NRC at 5 n.2. But "what may appear to raise significant hazards consideration at one time may, at some subsequent time and in light of technological advances and further study, be determined to present no significant hazards consideration." *See id.* at 6.

OPERATING LICENSE AMENDMENTS: NO SIGNIFICANT HAZARDS CONSIDERATION; CRITERIA FOR ISSUANCE

In adopting final implementing regulations for the Sholly Amendment, the Commission also provided in the Statement of Considerations particular criteria for making no significant hazards consideration determinations in spent fuel

amendment cases. *See* "Final Procedures and Standards on No Significant Hazards Considerations," 51 Fed. Reg. 7744, 7753-55 (Mar. 6, 1986).

MEMORANDUM AND ORDER

The Board of Commissioners of Orange County, North Carolina ("Orange County"), has filed a petition for review and request for immediate suspension and stay of the NRC Staff's no significant hazards consideration ("NSHC") determination and issuance of a license amendment for spent fuel pool expansion at the Shearon Harris nuclear power plant ("Shearon Harris"). Such a petition is not permitted by our regulations and we reject it summarily. However, to assist us in determining whether we should exercise our discretion and review the NRC Staff's NSHC determination in this specific case, we seek additional information and views from the Staff, and direct the Licensee temporarily not to store spent fuel under the license amendment pending further Commission order or a licensing board decision approving the amendment.

I. BACKGROUND

This proceeding involves a December 1998 license amendment application filed by Carolina Power & Light Company ("CP&L") to increase the spent fuel storage capacity at Shearon Harris. The Shearon Harris fuel handling building was originally designed and constructed with four separate storage pools to support four nuclear units. All four spent fuel pools had been constructed by the time three of the four Shearon Harris units were cancelled. Only pools A and B are currently in service. CP&L desires to add rack modules to spent fuel pools C and D and place pool C in service.

The Licensing Board granted Orange County intervenor status to challenge the application, and admitted two of Orange County's technical contentions. *See* LBP-99-25, 50 NRC 25 (1999). Following an oral argument held pursuant to 10 C.F.R. Part 2, Subpart K, the Board ruled that Orange County had presented no genuine and substantial dispute of fact or law requiring an evidentiary hearing and resolved the merits of the contentions in favor of CP&L. *See* LBP-00-12, 51 NRC 247 (2000). Orange County prematurely sought review of the Board's order while admissibility of the County's late-filed environmental contentions was pending, and the Commission denied the request for interlocutory review without prejudice. *See* CLI-00-11, 51 NRC 297 (2000).

The Board subsequently admitted one of Orange County's environmental contentions (EC-6) and heard oral argument on it on December 7, 2000, after receiving extensive written submissions from all parties. *See* LBP-00-19, 52 NRC 85 (2000). The Board has not yet issued its ruling on the environmental contention. The crux of the contention proposed by Orange County is whether a seven-step accident sequence, culminating in initiation of an exothermic oxidation reaction in spent fuel pools C and D,[1] has "a probability sufficient to provide the beyond-remote-and-speculative 'trigger' that is needed to compel preparation of an EIS [environmental impact statement] relative to [the] proposed licensing action." *See* 52 NRC at 95.

On December 21, 2000, the NRC Staff, pursuant to 10 C.F.R. §§ 50.58(b)(5) and 50.92, issued the license amendment, making it immediately effective on the ground that it raised no significant hazards consideration. The Staff is authorized by our rules to make such a determination

> if operation of the facility in accordance with the proposed amendment would not:
> (1) [i]nvolve a significant increase in the probability or consequences of an accident previously evaluated; or
> (2) [c]reate the possibility of a new or different kind of accident from any accident previously evaluated; or
> (3) [i]nvolve a significant reduction in a margin of safety.

See 10 C.F.R. § 50.92(c). Our rules implement a statutory directive, the so-called "Sholly Amendment," authorizing the Commission to issue immediately effective reactor license amendments, "in advance of the holding and completion of any required hearing," upon a "no significant hazards consideration" determination. *See* Atomic Energy Act § 189a(2)(A), 42 U.S.C. § 2239(a)(2)(A).

On December 22, 2000, Orange County submitted a "petition for review and request for immediate suspension and stay of the NRC staff's no significant hazards determination and issuance of license amendment for Harris spent fuel pool expansion" ("Orange County's Petition"). Orange County alleged that the no significant hazards consideration determination fails to satisfy the criteria in 10 C.F.R. § 50.92 and violates the National Environmental Policy Act ("NEPA"). *See* 42 U.S.C. §§ 4321-4347 (2000). A series of motions, responses, and replies ensued. The most recent was the NRC Staff's January 19 opposition to Orange County's motion to file a reply.

[1] The seven-step sequence is as follows: (1) a degraded core accident; (2) containment failure or bypass; (3) loss of all spent fuel cooling and makeup systems; (4) extreme radiation doses precluding personnel access; (5) inability to restart any pool cooling or makeup systems due to extreme radiation doses; (6) loss of most or all pool water through evaporation; and (7) initiation of an exothermic oxidation reaction in pools C and D.

II. DISCUSSION

A. Orange County's Petition

Our regulations provide that "[n]o petition or other request for review of or hearing on the staff's significant hazards consideration determination will be entertained by the Commission." *See* 10 C.F.R. § 50.58(b)(6). The regulations are quite clear in this regard. Accordingly, we reject Orange County's petition.

B. Exercise of the Commission's Discretion

Under our regulations, the Staff's determination on the no significant hazards consideration ("NSHC") issue is final, "subject only to the Commission's discretion, on its own initiative, to review the determination." *See id.* The Commission has inherent authority to exercise its discretionary supervisory authority to stay the Staff's actions or rescind the license amendment. *See Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-86-12, 24 NRC 1, 4-5 (1986), *rev'd and remanded on other grounds, San Luis Obispo Mothers for Peace v. NRC,* 799 F.2d 1268 (9th Cir. 1986).

In enacting the "Sholly Amendment" in 1983, Congress expressed "special concerns about significant hazards considerations for spent fuel license amendments." *Diablo Canyon,* CLI-86-12, 24 NRC at 5 n.2. But "what may appear to raise significant hazards consideration at one time may, at some subsequent time and in light of technological advances and further study, be determined to present no significant hazards consideration." *Id.* at 6. The Commission, in response to the stated congressional concern for spent fuel pools, obtained comment, used an outside contractor evaluation, and considered Staff recommendations on how to apply proposed rule criteria to spent fuel pool amendment cases. As a result of this process, in adopting final implementing regulations for the Sholly Amendment, the Commission also provided in the Statement of Considerations particular criteria for making NSHC determinations in spent fuel amendment cases.[2]

The NRC Staff's final NSHC determination and assessment of comments in this particular case do not appear to reference explicitly the specific spent fuel pool criteria as such, although information relevant to the criteria is identifiable in the SER. The Staff's NSHC determination also does not explicitly reference one of Orange County's comments on the then-proposed NSHC finding — i.e., the County's expression of concern about a severe accident scenario (whose probability the Licensing Board currently is assessing under the rubric of NEPA).

[2] *See* "Final Procedures and Standards on No Significant Hazards Considerations," 51 Fed. Reg. 7744, 7753-55 (Mar. 6, 1986).

Before deciding whether the Staff's NSHC determination requires further action by the Commission under its discretionary powers, therefore, we request additional information and views from the NRC Staff. Accordingly, we direct the Staff, within 14 days of the date of this Order, to file a brief addressing the 1986 NSHC criteria, the severe accident question, and any other aspect of the NSHC determination that, in the Staff's judgment, would benefit from elaboration. The Commission would be particularly interested in a summary of any quantitative data that underlie the Staff's NSHC determinations on accident probability, accident consequences, and margins of safety. Thus far, the Staff understandably has taken the position that it need file no merits pleading, as Orange County's petition for Commission review was unauthorized by our rules. Both Orange County and CP&L already have filed substantive briefs on the no significant hazards consideration issue. We will entertain no further filings on this issue from any party other than the NRC Staff.

To preserve the status quo while we consider the Staff's brief, we direct CP&L to store no spent fuel under the license amendment, pending a further order of the Commission or a licensing board decision approving the amendment, whichever comes sooner. *See* 10 C.F.R. § 2.764. CP&L may continue necessary prestorage activities should it so choose.

IT IS SO ORDERED.

For the Commission[3]

ANDREW L. BATES for
ANNETTE L. VIETTI-COOK
Secretary of the Commission

Dated at Rockville, Maryland,
this 14th day of February 2001.

[3] Commissioner Diaz was not present at the affirmation of this Order. Had he been present, he would have affirmed the prior vote to approve this Order.

Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated December 16, 1999, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 17th day of December 1999.

For the Nuclear Regulatory Commission.

Robert J. Fretz,

Project Manager, Section 1, Project Directorate IV & Decommissioning, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–33134 Filed 12–20–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–400]

Carolina Power & Light Company; Shearon Harris Nuclear Power Plant, Unit 1, Environmental Assessment and Finding of No Significant Impact

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF–63, issued to Carolina Power & Light Company (CP&L, the licensee), for operation of the Shearon Harris Nuclear Power Plant, Unit 1, (HNP) located in Wake and Chatham Counties, North Carolina.

Environmental Assessment

Identification of the Proposed Action

The proposed action would support a modification to HNP to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) 'C' and 'D' and placing the pools in service. The proposed action consists of: (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'; (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; and (3) an unreviewed safety question for additional heat load on the CCW system.

The proposed action is in accordance with the licensee's application for amendment dated December 23, 1998, as supplemented by letters dated April 30, June 14, July 23, September 3, October 15, and October 29, 1999.

The Need for the Proposed Action

The proposed action is needed for the licensee to provide spent fuel storage capacity for all four CP&L nuclear units (Harris, Brunswick 1 and 2, and Robinson) through the end of their current licenses.

HNP was originally planned as a four nuclear unit site and the fuel handling building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. HNP Units 3 and 4 were canceled in late 1981 and HNP Unit 2 was canceled in late 1983. The FHB, all four pools (including liners), and the cooling and cleanup system to support SFPs 'A' and 'B' were completed. However, construction on SFPs 'C' and 'D' was discontinued after Unit 2 was canceled and the system was not completed. HNP, Unit 1 began operation in 1987 with SFPs 'A' and 'B' in service.

As permitted by the HNP operating license issued on January 12, 1987, CP&L has implemented a spent fuel shipping program. Spent fuel from Brunswick (2 BWR units) and Robinson (1 PWR unit) is shipped to HNP for storage in the HNP SFPs. 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, it will be necessary to activate SFPs 'C' and 'D' and the associated cooling and cleanup system by early in the year 2000. Activation of these pools will provide spent fuel storage capacity for all four CP&L units through the end of their current operating licenses.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed action and concludes there are no significant environmental impacts. The factors considered in this determination are discussed below.

Radioactive Waste Treatment

HNP uses waste treatment systems designed to collect and process gaseous, liquid, and solid waste that might contain radioactive material. These radioactive waste treatment systems are discussed in the Final Environmental Statement (FES, NUREG–0972) dated October 1983, and evaluated in the Safety Evaluation Report (SER, NUREG–1083) dated November 1983. The proposal to increase the spent fuel storage capacity at HNP will not involve any change in the waste treatment systems described in the FES or SER.

Gaseous Radioactive Wastes

Gaseous releases from the fuel storage area are combined with other plant exhausts. Normally, the contribution from the fuel storage area is negligible compared to the other releases and no significant increases are expected as a result of the expanded storage capacity. Storing spent fuel in four pools (instead of the previous two pools) will result in an increase in the SFP evaporation rate. The licensee has determined that the increased evaporation will increase the relative humidity of the fuel building atmosphere by less than 10%. This increase is within the capacity of both the normal and the Engineered Safety Feature (ESF) ventilation systems. The net result of the increased heat loss and water vapor emission to the environment will be negligible.

Solid Radioactive Wastes

Spent resins are generated by the processing of SFP water through the SFP purification system. These spent resins are disposed of as solid radioactive waste. The necessity for pool filtration resin replacement is determined primarily by the requirement for water clarity, and the resin is normally expected to be changed about once a year. The licensee does not expect the resin change-out frequency of the SFP purification system to be permanently increased as a result of the expanded storage capacity. During racking operations, a small amount of additional resins may be generated by the pool cleanup system on a one-time basis.

Radiological Impact Assessment

For this modification the licensee plans to install region 2 (non-flux trap style) rack modules in pools 'C' and 'D' in incremental phases, on an as-needed basis. The licensee estimates that the collective dose associated with the proposed fuel rack installation is in the range of 2–3 person-rem.

All of the operations involved in racking will use detailed procedures prepared with full consideration of ALARA (as low as reasonably achievable) principles. The HNP racking

project represents low radiological risk because the pools currently contain no spent fuel. The Radiation Protection Department will prepare Radiation Work Permits (RWPs) for the various jobs associated with the SFP rack installation operation. These RWPs will instruct the project personnel in the areas of protective clothing, general dose rates, contamination levels and dosimetry requirements. Personnel will wear protective clothing and will be required to wear personnel monitoring equipment including alarming dosimeters.

Since the proposed license amendment does not involve the removal of any spent fuel racks, the licensee does not plan on using divers for this project. However, if it becomes necessary to use divers to remove any interferences which may impede the installation of the new spent fuel racks, the licensee will equip each diver with the appropriate monitoring equipment. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposure is maintained ALARA.

On the basis of its review of the HNP proposal, the staff concludes that the increase in spent fuel storage capacity at HNP can be accomplished in a manner that will ensure that doses to workers will be maintained ALARA.

Accident Considerations

In its application, the licensee evaluated the possible consequences of fuel handling accidents to determine offsite doses. The proposed SFP rack installation at HNP will not affect any of the assumptions or inputs used in evaluating the dose consequences of a fuel handling accident and, therefore, will not result in an increase in the doses from a postulated fuel handling accident. The proposed action will not change the procedures or equipment used for, or the frequency of, fuel moves at HNP or fuel shipments from the Brunswick and Robinson plants. Therefore, the probability of a postulated fuel handling accident will not increase from that previously evaluated.

The staff has previously considered accidents whose consequences might exceed a fuel handling accident; that is, beyond design basis events. One such accident evaluated by the staff involves a structural failure of the SFP, resulting in loss of all contained cooling water followed by heatup and a zirconium cladding fire. The details of this severe accident are discussed in NUREG/CR-4982, entitled "Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82." The staff also issued NUREG/CR-5176, entitled "Seismic Failure and Cask Drop Analysis of the Spent Fuel Pools at Two Representative Nuclear Power Plants." This report considers the structural integrity of the SFP and the pool response to the circumstances considered. Subsequently, the staff issued NUREG/CR-5281, "Value/Impact Analysis of Accident Preventative and Mitigative Options for Spent Fuel Pools," and NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82: Beyond Design Basis Accidents in Spent Fuel Pools." In NUREG-1353, the staff determined that no new regulatory requirements were warranted in relation to Generic Issue 82.

The staff believes that the probability of severe structural damage occurring at HNP is extremely low. This belief is based upon the Commission's requirements for the design and construction of SFPs and their contents and on the licensee's adherence to approved industry codes and standards. For example, in the HNP case, the pools are an integral part of the fuel building. The SFPs and the spent fuel storage racks are Seismic Category 1, and thus, are required to remain functional during and after a safe shutdown earthquake. In the unlikely event of a total loss of the cooling system, makeup water sources are available to replace coolant lost through evaporation or boiling. Therefore, the staff concludes that the potential for environmental impact from severe accidents is negligible.

The proposed action will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure. Therefore, there are no significant radiological environmental impacts associated with the proposed action.

With regard to potential nonradiological impacts, the proposed action does not involve any historic sites. It does not affect nonradiological plant effluents and has no other environmental impact. Therefore, there are no significant nonradiological environmental impacts associated with the proposed action.

Accordingly, the staff concludes that there are no significant environmental impacts associated with the proposed action.

Alternatives to the Proposed Action

A "Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel," NUREG-0575, Volumes 1-3, was issued by the Commission in August 1979. The finding of the FGEIS is that the environmental costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. The storage of spent fuel, as evaluated in NUREG-0575, is considered to be an interim action, not a final solution to permanent disposal. One spent fuel storage alternative considered in detail in the FGEIS is the expansion of the onsite fuel storage capacity by modification of the existing SFPs. The Commission has approved numerous applications for SFP expansion. The finding in each has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage design and limitations caused by spent fuel already stored in the pools, the FGEIS recommended that licensing reviews be done on a case-by-case basis, to resolve plant-specific concerns.

Specific alternatives to the proposed action are discussed below.

Shipment of Fuel to a Permanent Federal Fuel Storage/Disposal Facility

Shipment of spent fuel to a high-level radioactive storage facility is an alternative to increasing the onsite spent fuel storage capacity. However, DOE's high-level radioactive waste repository is not expected to begin receiving spent fuel until approximately 2010, at the earliest. In October 1996, the Administration did commit DOE to begin storing wastes at a centralized location by January 31, 1998. However, no location has been identified and an interim federal storage facility has yet to be identified in advance of a decision on a permanent repository. Therefore, shipping spent fuel to the DOE repository is not considered an alternative to increased onsite spent fuel storage capacity at this time.

Shipment of Fuel to a Reprocessing Facility

Reprocessing of spent fuel from HNP is not a viable alternative since there are no operating commercial reprocessing facilities in the United States. Therefore, spent fuel would have to be shipped to an overseas facility for reprocessing. However, this approach has never been used and it would require approval by the Department of State as well as other entities. Additionally, the cost of spent fuel reprocessing is not offset by the salvage value of the residual uranium; reprocessing represents an added cost. Therefore, this alternative is considered unacceptable.

Reduction of Spent Fuel Generation

Improved usage of fuel and/or operation at a reduced power level would decrease the amount of fuel being stored in the pool and thus increase the amount of time before full core off-load capability is lost. With extended burnup of fuel assemblies, the fuel cycle would be extended and fewer offloads would be necessary. The licensee has already increased its fuel enrichment to 5 percent and is currently operating on 18-month refueling cycles. Operating the plant at a reduced power level would not make effective use of available resources, and would cause unnecessary economic hardship on CP&L and its customers. Therefore, reducing the amount of spent fuel generated by increasing burnup further or reducing power is not considered a practical alternative.

Alternative Creation of Additional Storage Capacity

Alternative technologies that would create additional storage capacity include rod consolidation, dry cask storage, and modular vault dry storage. Rod consolidation involves disassembling the spent fuel assemblies and storing the fuel rods from two or more assemblies in a stainless steel canister that can be stored in the spent fuel racks. Industry experience with rod consolidation is currently limited, primarily due to concerns for potential gap activity release due to rod breakage, the potential for increased fuel cladding corrosion due to some of the protective oxide layer being scraped off, and because the prolonged consolidation activity could interfere with ongoing plant operations. Dry cask storage is a method of transferring spent fuel, after storage in the pool for several years, to high capacity casks with passive heat dissipation features. After loading, the casks are stored outdoors on a seismically qualified concrete pad. Concerns for dry cask storage include the potential for fuel or cask handling accidents, potential fuel clad rupture due to high temperatures, increased land use, construction impacts, the need for additional security provisions, and high costs. Vault storage consists of storing spent fuel in shielded stainless steel cylinders in a horizontal configuration in a reinforced concrete vault. The concrete vault provides missile and earthquake protection and radiation shielding. Concerns for vault dry storage include the need for additional security provisions, increased land use, construction impacts, eventual decommissioning of the new vault, the potential for fuel or clad rupture due to high temperatures, and high cost.

The environmental impacts of the alternative technologies discussed above and the proposed action are similar.

The No-Action Alternative

As an alternative to the proposed action, the staff also considered denial of the proposed action (i.e., the "no-action" alternative). Denial of the application would result in no change in current environmental impacts.

Alternative Use of Resources

This action does not involve the use of any resources not previously considered in the Final Environmental Statement for HNP.

Agencies and Persons Consulted

In accordance with its stated policy, on December 2 and 3, 1999, the staff consulted with North Carolina State officials, Mr. Richard M. Fry and Mr. Johnny James of the North Carolina Department of Environment and Natural Resources, regarding the environmental impact of the proposed action. The State officials stated that they had no objection to the finding. However, they requested that the staff hold a public meeting in Raleigh, North Carolina to discuss the license amendment review process, the results of the review for HNP's proposed amendment, and the analysis that led to this environmental assessment finding.

Finding of No Significant Impact

On the basis of the environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the proposed action.

For further details with respect to the proposed action, see the licensee's letter dated December 23, 1998, as supplemented by letters dated April 30, June 14, July 23, September 3, October 15, and October 29, 1999, which are available for public inspection at the Commission's Public Document Room, The Gelman Building, 2120 L Street, NW., Washington, DC.

Dated at Rockville, Maryland, this 15th day of December 1999.

For the Nuclear Regulatory Commission.

Richard P. Correia,

Chief, Section 2, Project Directorate II, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 99–33023 Filed 12–20–99; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 50–269, 50–270, and 50–287]

Duke Power Company; Notice of Availability of the Final Supplement 2 to the Generic Environmental Impact Statement for the License Renewal of Oconee Nuclear Station, Unit Nos. 1, 2, and 3

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has published a final plant-specific Supplement 2 to the Generic Environmental Impact Statement for License Renewal of Nuclear Plants (GEIS) (NUREG–1437) regarding the renewal of operating licenses DPR–38, DPR–47, and DPR–55 for an additional 20 years of operation at the Oconee Nuclear Station (ONS) Units 1, 2, and 3, respectively. ONS is located in Oconee County, South Carolina. Possible alternatives to the proposed action (license renewal) include no action and reasonable alternative energy sources.

In Section 9.3 of the report, the staff concludes:

> Based on (1) the analysis and findings in the Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants, NUREG–1437, (2) the ER [Environmental Report] submitted by Duke, (3) consultation with other Federal, State, and local agencies, (4) the staff's own independent review, and (5) the staff's consideration of public comments, the staff recommends that the Commission determine that the adverse environmental impacts of license renewal for Oconee Nuclear Station, Units 1, 2, and 3 are not so great that preserving the option of license renewal for energy planning decisionmakers would be unreasonable.

The final supplement to the GEIS for ONS is available for public inspection and copying at the Commission's Public Document Room at the Gelman Building, 2120 L Street NW., Washington, DC.

FOR FURTHER INFORMATION, CONTACT: Mr. James H. Wilson, Generic Issues, Environmental, Financial, and Rulemaking Branch, Division of Regulatory Improvement Programs, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Mr. Wilson can be contacted at (301) 415–1108 or by writing to: James H. Wilson, U.S. Nuclear Regulatory Commission, MS 0–11 F1, Washington, DC 20555.

Dated at Rockville, Maryland, this 9th day of December 1999.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:

G. Paul Bollwerk, III, Chairman
Frederick J. Shon
Dr. Peter S. Lam

In the Matter of **Docket No. 50-400-LA**
(ASLBP No. 99-762-02-LA)

CAROLINA POWER & LIGHT COMPANY
(Shearon Harris Nuclear Power Plant) **July 12, 1999**

In this proceeding concerning Applicant Carolina Power and Light Company's (CP&L) request to increase the spent fuel storage capacity of its Shearon Harris Nuclear Power Plant through a 10 C.F.R. § 50.90 facility operating license amendment, the Licensing Board grants the hearing request of the Board of Commissioners of Orange County, North Carolina (BCOC), concluding BCOC has standing and has proffered two admissible contentions challenging CP&L's proposed fuel storage expansion plan.

RULES OF PRACTICE: STANDING TO INTERVENE

Those who seek party status in NRC adjudicatory proceedings must demonstrate that they fulfill the contemporaneous judicial standards for standing, which require that a participant establish: (1) it has suffered or will suffer a distinct and palpable injury that constitutes injury in fact within the zone of interests arguably protected by the governing statutes (e.g., the Atomic Energy Act of 1954 (AEA), the National Environmental Policy Act of 1969 (NEPA)); (2) the injury is fairly traceable to the

challenged action; and (3) injury is likely to be redressed by a favorable decision. *See Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-1, 43 NRC 1, 6 (1996).

RULES OF PRACTICE: STANDING TO INTERVENE (ZONE OF INTERESTS; REDRESSABILITY OF INJURIES)

The safety and environmental concerns alleged by a local governmental organization relative to its citizens and their local habitat fall within the statutory zone of interests implicated in this proceeding and those injuries could be redressed by a favorable decision in this proceeding.

RULES OF PRACTICE: STANDING TO INTERVENE (ORGANIZATIONAL)

As the Commission has recognized in a somewhat different context, the strong interest that a governmental body has in protecting the individuals and territory that fall under its sovereign guardianship establishes an organizational interest for standing purposes. *See Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-98-13, 48 NRC 26, 33 (1998).

RULES OF PRACTICE: STANDING TO INTERVENE (INJURY IN FACT; FACTUAL REPRESENTATION)

During the threshold standing inquiry, a petitioner need not establish an asserted injury in fact basis for assertions of offsite radiological consequences with "certainty" or provide extensive technical studies. *See Sequoyah Fuels Corp.* (Gore, Oklahoma Site), CLI-94-12, 40 NRC 64, 72 (1994). Such an assertion of injury in fact will be accepted if it is at least facially plausible that it is neither remote nor speculative and the opposing party fails to establish a fatal flaw in its analysis.

RULES OF PRACTICE: CONTENTIONS (POSSIBLE FAILURE TO COMPLY WITH REGULATORY REQUIREMENT)

In order to posit a contention that requires the analysis of an action violating a specific technical specification, a petitioner would have to make some particularized demonstration that there is a reasonable basis to believe that the applicant will act contrary to the terms of such a requirement. *See General Public Utilities Nuclear Corp.* (Oyster Creek Nuclear Generating Station), LBP-96-23, 44 NRC 143, 164 (1996).

RULES OF PRACTICE: HEARING PROCEDURES FOR SPENT FUEL POOL EXPANSION PROCEEDING

A spent fuel capacity expansion proceeding is subject to the hybrid hearing process outlined in 10 C.F.R. Part 2, Subpart K, to the degree that any party wishes to invoke those procedures. Any party that wishes to invoke this process must do so within 10 days of an order granting a hearing request. *See* 10 C.F.R. § 2.1109(a)(1). If invoked, the process would consist of the following: a 90-day discovery period followed by the simultaneous written submission of relevant facts, data, and arguments and an oral argument on the issue whether an evidentiary proceeding is required for any of the contentions; and finally a decision by the presiding officer that both designates disputed issues of fact for an evidentiary hearing and resolves any other issues. *See* 10 C.F.R. §§ 2.1111, 2.1113(a), 2.1115(a)-(b).

MEMORANDUM AND ORDER
(Ruling on Standing and Contentions)

Responding to a January 7, 1999 notice of opportunity for a hearing, 64 Fed. Reg. 2237 (1999), Petitioner Board of Commissioners of Orange County, North Carolina (BCOC), has filed a timely hearing request and intervention petition that is now before the Board. In its February 12, 1999 petition, BCOC challenges the December 23, 1998 request of Applicant Carolina Power & Light Co. (CP&L) for permission to increase the spent fuel storage capacity at its Shearon Harris Nuclear Power Plant (Harris), which is located in Wake and Chatham Counties, North Carolina. If granted, CP&L's 10 C.F.R. § 50.90 facility operating license amendment request would permit it to add rack modules to spent fuel pools C and D and place those pools in operation.

Both the Applicant and the NRC Staff have contested the BCOC request. CP&L asserts that BCOC lacks standing to intervene, while both CP&L and the Staff argue that none of BCOC's eight contentions are admissible. Having concluded that BCOC does have standing and has proffered two admissible contentions, for the reasons set forth below we grant its hearing request.

I. BACKGROUND

In its December 1998 license amendment request, CP&L indicated that the fuel handling building (FHB) at the Harris site was originally designed and constructed with four separate spent fuel pools to accommodate the four reactor units that were planned for the site. Pools A through D were anticipated to serve Units 1 through

4, respectively. Although three of the units were canceled in the early 1980s, the FHB, the four pools (with liners), and the cooling and cleanup system to support pools A and B were completed and turned over to CP&L. Construction on the cooling and cleanup system for pools C and D, however, was not completed. CP&L also declared that because a Department of Energy high-level waste repository is not expected to be available in the foreseeable future, it has been shipping spent fuel from its three other nuclear facilities for storage in the Harris pools in order to maintain full core offload capability for those facilities. According to CP&L, the present amendment request to utilize pools C and D is designed to provide storage capacity for all four CP&L units — Harris, Brunswick Steam Electric Plant, Units 1 and 2, and H.B. Robinson, Unit 2 — through the end of their current operating licenses. *See* CP&L Request for License Amendment (Dec. 23, 1998) Encl. 1, at 1 [hereinafter License Amendment].

Asserting it had standing to intervene on behalf of its citizens, in its February 12, 1999 intervention petition BCOC contested this CP&L request as involving both safety and environmental risks. *See* [BCOC] Request for Hearing and Petition to Intervene (Feb. 12, 1999) at 2-4 [hereinafter BCOC Petition]. CP&L filed a March 1, 1999 answer declaring that the BCOC petition to intervene should be denied because BCOC has failed to establish its standing. *See* [CP&L] Answer to BCOC's Request for Hearing and Petition to Intervene (Mar. 1, 1999) at 7-11 [hereinafter CP&L Petition Response]. The NRC Staff, on the other hand, asserted in its answer that BCOC had established its standing to intervene. *See* NRC Staff's Answer to Orange County's Request for Hearing and Petition to Intervene (Mar. 4, 1999) at 5 [hereinafter Staff Petition Response].

In its initial prehearing order, the Board set an April 5, 1999 deadline for BCOC to submit a supplement to its petition specifying its contentions. *See* Licensing Board Memorandum and Order (Initial Prehearing Order) (Feb. 24, 1999) at 3 (unpublished). BCOC filed a supplemental petition on that date, which set forth three technical and five environmental contentions. *See* [BCOC] Supplemental Petition to Intervene (Apr. 5, 1999) at 4-44 [hereinafter BCOC Contentions]. In responses filed May 5, 1999, both CP&L and the Staff took the position that BCOC had failed to present a contention that would meet the admissibility standards set forth in 10 C.F.R. § 2.714(b) and, as such, its petition should be dismissed. *See* [CP&L] Answer to Petitioner [BCOC] Contentions (May 5, 1999) [hereinafter CP&L Contentions Response]; NRC Staff's Response to [BCOC] Supplemental Petition to Intervene (May 5, 1999) [hereinafter Staff Contentions Response]. Thereafter, at a one-day prehearing conference conducted in Chapel Hill, North Carolina, on May 13, 1999, the Board heard oral arguments from the participants on the issues of BCOC's standing and the admissibility of its eight contentions. *See* Tr. at 11-170.

II. ANALYSIS

A. Standing

Those who seek party status in NRC adjudicatory proceedings must demonstrate that they fulfill the contemporaneous judicial standards for standing, which require that a participant establish (1) it has suffered or will suffer a distinct and palpable injury that constitutes injury-in-fact within the zone of interests arguably protected by the governing statutes (e.g., the Atomic Energy Act of 1954 (AEA), the National Environmental Policy Act of 1969 (NEPA)); (2) the injury is fairly traceable to the challenged action; and (3) injury is likely to be redressed by a favorable decision. *See Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), CLI-96-1, 43 NRC 1, 6 (1996).

In this instance, BCOC asserts in its intervention petition that, as a political subdivision of the State of North Carolina, it is "authorized to protect the citizens of the County through its police powers," and indicates it wishes to intervene because the proposed spent fuel pool expansion amendment "threatens the County's interest in protecting the health and welfare of its citizens and the integrity of the environment in which they live." BCOC Petition at 3; *see also* Tr. at 12. BCOC also declares that "[t]he entire county lies within the 50-mile ingestion exposure emergency planning zone around the Harris facility, and part of the county lies within 15 miles of the plant." BCOC Petition at 3. According to BCOC, in light of the showing in the attachments to its petition regarding the increased risk of, and offsite consequences resulting from, reactor or spent fuel pool accidents that could occur if the CP&L expansion proposal is implemented, it has demonstrated its injury in fact. *See* Tr. at 12-15. The Staff agrees that BCOC has made a showing sufficient to establish BCOC's organizational standing. *See* Staff Petition Response at 5 & n.2. CP&L objects, however, declaring that BCOC — which CP&L maintains is located approximately 17 miles from the Harris facility — has not established its organizational standing. *See* CP&L Petition Response at 7-8; Tr. at 15-21.

It is apparent that the safety and environmental concerns alleged by BCOC fall within the statutory zone of interests implicated in this proceeding and that those injuries could be redressed by a favorable decision in this proceeding. Moreover as the Commission has recognized in a somewhat different context, the strong interest that a governmental body like BCOC has in protecting the individuals and territory that fall under its sovereign guardianship establishes an organizational interest for standing purposes. *See Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-98-13, 48 NRC 26, 33 (1998).

Indeed, there seems little doubt that if the Harris facility were located within the boundaries of Orange County, the requisite injury in fact would have been established relative to Petitioner BCOC. *See Private Fuel Storage, L.L.C.*

(Independent Spent Fuel Storage Installation), LBP-98-7, 47 NRC 142, 169 (finding State of Utah has standing relative to facility located within the State, albeit on Native American reservation), *aff'd on other grounds,* CLI-98-13, 48 NRC 26 (1998). It is not so located, however. Instead, the county's closest boundary is approximately 17 miles from the facility. Previous standing rulings regarding spent fuel pool expansion and reracking indicate that standing has been accorded to interested persons within approximately 10 miles of the reactor facility.[1] *See Florida Power & Light Co.* (St. Lucie Nuclear Power Plant, Unit 1), LBP-88-10A, 27 NRC 452, 455, *aff'd,* ALAB-893, 27 NRC 627 (1988); *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), LBP-87-17, 25 NRC 838, 842, *aff'd in part and rev'd in part on other grounds,* ALAB-869, 26 NRC 13 (1987); *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), LBP-87-7, 25 NRC 116, 118 (1987). While CP&L declares that the additional 7 miles to the BCOC border negates BCOC's standing claim, we conclude the additional distance is not a bar to Petitioner's standing in this instance.

In an affidavit attached both to BCOC's petition and its contentions supplement, Dr. Gordon Thompson, the executive director of the Institute for Resource and Security Studies, analyzes the hazard posed by the Harris spent fuel pool expansion as it relates to cesium-137.[2] Noting that cesium-137 is an important hazard potential indicator because it emits intense gamma radiation and is released comparatively readily in severe accidents, Dr. Gordon declares that activation of pools C and D will potentially result in an inventory of spent fuel containing cesium-137 in amounts that, if released in a significant fraction to the environment because of a severe accident, would create offsite radiation doses in amounts that would be an order of magnitude larger than the exposure from the Chernobyl accident and as much as two times higher than those from a similar accident involving only pools A and B. He also notes that, as is the case with many facilities, the spent fuel pools at the Harris plant are not within the containment area, so that any released radioisotopes are likely to exit the building in an atmospheric plume. He further postulates what he asserts are the previously unanalyzed consequences of a partial uncovering of the fuel, which he declares could be more severe than the total water loss circumstances previously analyzed in terms of the possibility of creating exothermic reactions that could result in significant atmospheric discharges. Finally, he identifies several events involving the pools

[1] In addition to the cases cited above, in *Virginia Electric and Power Co.* (North Anna Nuclear Power Station, Units 1 and 2), ALAB-522, 9 NRC 54, 55-57 (1979), the Appeal Board permitted intervention in a spent fuel pool expansion proceeding for an intervenor group that had identified members who resided 35 and 45 miles from the facility, one of whom also engaged in canoeing on a river "in the general vicinity" of the plant. Although the exact basis for this ruling is not entirely clear, because it appears to rest on the close proximity of the recreational activities to the facility rather than the more remote residences of the individuals, we do not consider it controlling here.

[2] This attachment was originally prepared to support a challenge to the Staff's proposed no significant hazards consideration finding that accompanied the hearing opportunity notice for the CP&L amendment. The validity of that proposed determination is, of course, not a matter before us. *See* 10 C.F.R. § 50.91(a)(4).

or an interaction between the pools and the Harris reactor, that might cause such a partial water loss accident. *See* BCOC Contentions, Exh. 2, at 6-10; *see also* BCOC Contentions at 29-32.

Relative to the standing criterion of injury in fact, what Dr. Thompson's declaration indicates is that the proposed CP&L expansion could create circumstances in which there could be releases that could go beyond the Harris facility boundary and could have health or environmental impacts equal to or in excess of those that now exist for pools A and B. CP&L, however, posits two reasons why this showing is insufficient to establish BCOC's standing. First, citing the Commission's decision in *Sequoyah Fuels Corp.* (Gore, Oklahoma Site), CLI-94-12, 40 NRC 64, 72 (1994), it argues that Dr. Thompson's analysis relies on beyond-design-basis accident sequences that are too conjectural or hypothetical to provide a basis for standing. *See* CP&L Petition Response at 10. In addition, it points out that the Staff recently has granted a series of exemptions waiving offsite emergency planning requirements for power reactor facilities that have been shut down, but will retain spent fuel inventories in pools during the decommissioning process. *See id* at 11 & n.8 (citing, as an example, 63 Fed. Reg. 48,768 (1998) (Maine Yankee exemption)); Tr, at 19.

We find neither of these arguments persuasive. The Commission indicated in *Sequoyah Fuels,* CLI-94-12, 40 NRC at 72, that during the threshold standing inquiry, a petitioner need not establish an asserted injury in fact basis with "certainty" or provide extensive technical studies. *Id.* Here, in conformance with that standard, BCOC has produced an explanation of why Dr. Thompson's accident concerns are not remote and speculative that is at least facially plausible. *See* BCOC Contentions at 31-32. At the same time, nothing presented by CP&L, including the referenced emergency planning exemptions, establishes a fatal flaw in his analysis. The exemptions involve facilities in which the power reactors are no longer operating, a crucial distinction given Dr. Thompson's specific reference to pool-reactor operation interaction as a supporting basis for his analysis.

Accordingly, we conclude that BCOC has made a showing sufficient to establish that it meets the criteria for standing in this proceeding.

B. Contentions

As was noted earlier, in seeking to gain party status to this proceeding, BCOC has proffered eight contentions, three involving technical issues and five that concern environmental matters. For reasons that will become apparent, we deal with the admissibility of the technical contentions individually, but rule on the environmental contentions as a group.

1. *Technical Contentions*[3]

TECHNICAL CONTENTION 1 (TC-1) — Inadequate Emergency Core Cooling and Residual Heat Removal

> CONTENTION: In order to cool spent fuel storage pools C and D, CP&L proposes to rely on the Unit 1 Component Cooling Water ("CCW") system, coupled with administrative measures to ensure that the heat load from the pools does not overtax the CCW system. CP&L's reliance on the Unit 1 CCW system and administrative measures for cooling spent fuel storage pools C and D will unduly compromise the effectiveness of the residual heat removal ("RHR") system and the Emergency Core Cooling System ("ECCS") for the Shearon Harris plant, such that the plant will not comply with Criteria 34 and 35 of Appendix A to 10 C.F.R. Part 50.

DISCUSSION: BCOC Contentions at 4-10; CP&L Contentions Response at 12-28; Staff Contentions Response at 4-10; Tr. at 29-87.

RULING: In discussing this contention, we utilize the six-basis construct outlined in the CP&L response to the BCOC contention supplement, which we find both useful and accurate.

> a. Basis 1 — Even without the amendment to add pools C and D, the Harris Final Safety Analysis Report (FSAR) shows that the CCW system is incapable of accommodating the heat load from the recirculation phase of a design-basis loss of coolant accident (LOCA).

Although it questions the adequacy of the existing CCW system, BCOC has failed to provide any factual information or expert opinion that gives us reason to believe the relatively small addition to the heat load during a LOCA would have any effect on the ability of the system to cool the reactor. CP&L presented figures in its contention response and at the prehearing conference indicating that the heat removal capabilities of the system are adequate. *See* CP&L Contentions Response at 16-17; Tr. at 56-57. Petitioner BCOC does not offer any specific calculation showing otherwise, nor did BCOC's expert allege that any specific limit would be violated. *See* Tr. at 34-39. The fact that BCOC's expert used an outdated version of the FSAR casts further doubt on the notion that any limits would be exceeded, and the Petitioner's difficulties in identifying the latest version of the FSAR, while unfortunate, cannot form the basis for a valid contention.

Accordingly, lacking adequate factual and expert opinion support, this basis is insufficient to support the contention. *See Private Fuel Storage,* LBP-98-7, 47 NRC at 180-81. In fact, in its present form, this basis appears to be a challenge to the design of the emergency core cooling system (ECCS), which would place

[3] Because we prefer to have these first three contentions designated by their subject matter category, i.e., technical, we have renumbered them as technical contentions 1 through 3.

it outside the scope of this proceeding, and so again does not provide support for an admissible contention. *See id.* at 179.

> b. Basis 2 — The analysis of CCW margin supporting the license amendment application does not address the time dependence of the CCW system heat load during a design-basis LOCA.

Basis 2, questioning the time dependence of the heat load analysis, likewise is without foundation. The short of it is that CP&L did indeed take account of the time variation, as both it and the Staff point out. *See* CP&L Contentions Response at 17-20; Staff Contentions Response at 6-7; Tr. at 63-65. Petitioner's plea that the time dependence is complex, *see* Tr. at 40, raises no litigable issue. No one doubts this issue is complex; however, an allegation of complexity is not a substitute for an adequately supported explanation of the exact nature of the matter in controversy. Nor is the BCOC complaint that some calculation sheets may not have been signed, *see id.*, adequate to call the substance of the calculations into question, as would be necessary for any cognizable challenge to their accuracy. Thus, besides problems with its materiality, this basis lacks sufficient factual and/or expert opinion support to make this a litigable issue. *See Private Fuel Storage,* LBP-98-7, 47 NRC at 179-81.

> c. Basis 3 — The analysis of CCW margin supporting the license amendment application does not address the degradation of CCW and RHR heat exchanger performance due to heat exchanger fouling and plugging.

TC-1, Basis 3, alleging a failure to account for fouling and plugging factors in the calculation of the analysis of the CCW margin, is simply incorrect. CP&L apparently did account for such factors, *see* CP&L Contentions Response at 20-22; Staff Contentions Response at 7, and the fact BCOC generally is dissatisfied with the level of detail in the calculation and is not sure whether the calculation has been finalized, *see* Tr. at 44, cannot form the basis of an admissible contention. *See Private Fuel Storage,* LBP-98-7, 47 NRC at 180-81.

> d. Basis 4 — The license amendment application does not address the potential for failure to comply with the administrative measure limiting the heat load in pools C and D to 1.0 MBTU/hour.

Basis 4, asserting an improper reliance on an administrative limit to keep the heat load in pools C and D within safe bounds, scarcely represents a change introduced by the proposed license amendment, as Petitioner would have us find. The heat load in existing pools A and B, and indeed many other limits, depend ultimately upon administrative controls. And there are many safety parameters like these administrative controls that could, at the discretion of the operating

organization, be pushed beyond their appropriate limits. That, however, is precisely the reason for the adoption of technical specifications.

Among other things, technical specifications are intended to prevent the licensee organization from exceeding a limit in a way that could pose a hazard. In the case of this license amendment, there is a proposed technical specification, Technical Specification 5.6.3.d, *see* License Amendment, Encl. 5, at unnumbered p. 4, that would dictate that the stored fuel heat load for pools C and D not exceed 1.0 MBtu/hr. Given this provision, we agree with CP&L and the Staff, and the Licensing Board's ruling in *General Public Utilities Nuclear Corp.* (Oyster Creek Nuclear Generating Station), LBP-96-23, 44 NRC 143, 164 (1996), that in order to posit a contention that requires the analysis of an action violating a specific technical specification, a petitioner would have to make some particularized demonstration that there is a reasonable basis to believe that the applicant will act contrary to the terms of such a requirement. Thus, in this instance, BCOC would need to show that circumstances exist that make the proposed technical specification especially prone to violation, which it has not done.

e. Basis 5 — The license amendment application does not address the potential for increased operator error in diverting CCW system flow to meet the cooling needs of pools C and D during a LOCA event.

Basis 5 lacks specificity, as well as failing to raise any issue that is directly related to the change proposed in the present amendment. In this regard, CP&L and the Staff have indicated that the added burden on the operators is vanishingly small; the requirement to restore pool cooling already exists (and, indeed, exists for pools A and B with their substantially greater heat load); and the failure to perform that minor function would not lead to a substantial hazard. *See* CP&L Contentions Response at 23-26; Staff Contentions Response at 8-9; Tr. at 69-71). In the face of this information, Petitioner's speculation that there may be excessive strains on the operators or that there may be critical temperature or humidity limits, *see* Tr. at 49-51, is simply that — speculation. Because BCOC has not identified any specific errors or hazards that may be occasioned or any specific limits that may be violated and has presented no calculations that can form the basis for this contention, it lacks adequate support. *See Private Fuel Storage,* LBP-98-7, 47 NRC at 180-81.

f. Basis 6 — The analysis supporting the license amendment application does not address the ability of Unit 1 electrical systems to meet the needs of pools C and D while also supporting essential safety functions.

Basis 6, a complaint that CP&L has failed to analyze the new demands on the emergency diesel generator system, also lacks adequate support. *See* CP&L Contentions Response at 26-28; Staff Contentions Response at 9. The analysis supporting the amendment indicates that the diesel generators have capacity to spare. *See* Tr. at 66-67. And Petitioner's additional plea that the time dependency of these loads may somehow show the system to be inadequate, *see* Tr. at 42, is again purely speculative. BCOC has given no reason to assume there is a time-dependent load that exceeds the peak given by CP&L in its analysis. *See Private Fuel Storage,* LBP-98-7, 47 NRC at 180-81.

In sum, we find TC-1 lacks an adequate basis and thus fails to meet the requirements for admissibility specified in 10 C.F.R. § 2.714(b).

TECHNICAL CONTENTION 2 (TC-2) — Inadequate Criticality Prevention

CONTENTION: Storage of pressurized water reactor ("PWR") spent fuel in pools C and D at the Harris plant, in the manner proposed in CP&L's license amendment application, would violate Criterion 62 of the General Design Criteria ("GDC") set forth in Part 50, Appendix A. GDC 62 requires that: "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." In violation of GDC 62, CP&L proposes to prevent criticality of PWR fuel in pools C and D by employing administrative measures which limit the combination of burnup and enrichment for PWR fuel assemblies that are placed in those pools. This proposed reliance on administrative measures rather than physical systems or processes is inconsistent with GDC 62.

DISCUSSION: BCOC Contentions at 10-13; CP&L Contentions Response at 29-36; Staff Contentions Response at 10-13; Tr. at 88-118.

RULING: In discussing this contention, we utilize CP&L's two-basis construct, which we again find both useful and accurate.

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

The Board has determined that this basis for the contention does indeed raise a genuine material dispute that warrants further inquiry so as to be cognizable in this proceeding. Specifically, the litigable issue essentially is a question of law: Does GDC 62 permit an applicant to take credit in criticality calculations for enrichment and burnup limits in fuel, limits that will ultimately be enforced by administrative controls?

While it is apparent that draft Regulatory Guide 1.13, at 1.13-13 to -15 (proposed rev. 2, Dec. 1981), *see* Staff Contentions Response, Attach. 3, would permit criticality control by such limits, the PFS-referenced Commission admonition that "[i]f there is conformance with regulatory guides, there is likely to be compliance with the GDC," *Petition for Emergency and Remedial Action,* CLI-78-6, 7 NRC 400, 407 (1978), is not a blanket endorsement of the notion that regulatory guides necessarily govern. Further, the instances cited by PFS in which the Staff issued licenses embodying administrative controls based on burnup and enrichment to

prevent criticality are instances that stand, to the extent they stand for anything, for the proposition that the Staff agrees with itself that its interpretation of this GDC is correct. The propriety of that interpretation of GDC 62 has apparently never been tested in the crucible of an adversary adjudication. We will permit such a test here by entertaining 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 requirements of GDC 62 for the prevention of criticality.

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

The second basis raises a question of fact: Will a single fuel assembly misplacement, involving a fuel element of the wrong burnup or enrichment, cause criticality in the fuel pool, or would more than one such misplacement or a misplacement coupled with some other error be needed to cause such criticality? While CP&L and the Staff both assure us that, when account is taken for the boron present in the fuel pool water, a single misplacement cannot lead to criticality, the fact that the Staff has sought further information on this point, as evidenced by exhibit 1 proffered by Orange County during the prehearing conference,[4] suggests that further inquiry on the validity of any calculations involved is warranted in determining whether the required single failure criterion is met. Clearly the nature of the amendment, introducing as it does the presence of high density racks on the site, involves a change that may call into question conformance with this aspect of the regulations. Accordingly, we admit contention TC-2 relative to this basis as well.

TECHNICAL CONTENTION 3 (TC-3) — Inadequate Quality Assurance[5]

CONTENTION: 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 the 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).

[4] While the pendency of a Staff requests for additional information (RAI) such as BCOC exhibit 1 is not a basis for delaying the filing of contentions, such an RAI may provide the basis for a contention. *See Baltimore Gas & Electric Co.* (Calvert Cliffs Nuclear Power Plant, Units 1 and 2), CLI-98-25, 48 NRC 325, 349-50 (1998), *petitions for review pending*, Nos. 99-1002 & 99-1043 (D.C. Cir. Jan. 4, 1999 & Feb. 8, 1999).

[5] The wording of this contention reflects the uncontested BCOC revision provided to the Board, *see* [BCOC] Response to [PFS] Proposed Rewording of Contention 3, Regarding Quality Assurance (May 27, 1999) at 2, with one Board clarification that is indicated by brackets.

Moreover, the Alternative 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 the remote camera, and the weld quality cannot be demonstrated adequately by circumstantial evidence.

DISCUSSION: BCOC Contentions at 13-19; CP&L Contentions Response at 36-48; Staff Contentions Response at 13-16; Tr. at 118-53.

RULING: We also will admit contention TC-3 for litigation. First, it is unclear from the present filings whether the criteria of Appendix B are to be enforced or not. CP&L says they will be complied with. *See* CP&L Contentions Response at 40. The Staff says they need not be. *See* Staff Contentions Response at 15. BCOC clearly believes they must be met. If, indeed, the criteria here applicable are those of 10 C.F.R. § 50.55a(a)(3), they require the Applicant to demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or
(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Such criteria are inherently more nebulous and governed by subjective judgment to a greater degree than those otherwise applicable to quality assurance matters under 10 C.F.R. Part 50, App. B, and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. In particular, we have heard nothing about such points as "hardship," "difficulty," or "compensating increase in the level of quality and safety." And, 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, a question about which there is clearly a dispute between CP&L and BCOC.

It also is clear from the positions of all the participants that some of the piping and equipment have not been properly stored and proper records regarding its quality during that period have not been maintained. Whether such storage and maintenance are necessary as a matter of law and fact is clearly a subject of dispute among the participants. The argument concerning this point is not a simple one nor do we have material on which we can rely to determine the matter.

We are presently uncertain as to the exact scope of the failure to meet the requirements of the regulations, and that scope is uncertain concerning both the

equipment involved and the extent to which each piece of equipment may itself be lacking. Although we heard participant presentations on these matters, much of this bordered on testimony submitted without the purifying challenge of cross examination by parties familiar with the details through discovery.

Thus, to recap, contention TC-1 is rejected as inadmissible while contentions TC-2 and TC-3 are accepted for litigation in the form and subject to the interpretations set forth above.

2. *Environmental Contentions*[6]

Petitioner BCOC specified five environmental contentions in its supplement, as follows:

ENVIRONMENTAL CONTENTION 1 (EC-1) — Proposed License Amendment Not Exempt from NEPA

CONTENTION: CP&L errs in claiming that the proposed license amendment is exempt from NEPA under 10 C.F.R. § 51.22.

ENVIRONMENTAL CONTENTION 2 (EC-2) — Environmental Impact Statement Required

CONTENTION: The proposed license amendment is not supported by an Environmental Impact Statement ("EIS"), in violation of NEPA and NRC's implementing regulations. An EIS should examine the effects of the proposed license amendment on the probability and consequences of accidents at the Harris plant. As required by NEPA and Commission policy, it should also examine the costs and benefits of the proposed action in comparison to various alternatives, including Severe Accident Design Mitigation Alternatives and the alternative of dry cask storage.

ENVIRONMENTAL CONTENTION 3 (EC-3) — Scope of EIS Should Include Brunswick and Robinson Storage

CONTENTION: The EIS for the proposed license amendment should include within its scope the storage of spent fuel from the Brunswick and Robinson nuclear power plants.

ENVIRONMENTAL CONTENTION 4 (EC-4) — Even if No EIS Required, Environmental Assessment Required

CONTENTION: Even if the Licensing Board finds that no EIS is required, it must order the preparation of an EA.

[6] BCOC numbered these contentions sequentially as contentions 4 through 8. As with the technical contentions, we prefer to see them designated by their subject matter category, i.e., environmental, and so renumber them accordingly.

ENVIRONMENTAL CONTENTION 5 (EC-5) — Discretionary EIS Warranted

CONTENTION: Even if the Licensing Board determines that an EIS is not required under NEPA and 10 C.F.R. § 51.20(a), the Board should nevertheless require an EIS as an exercise of its discretion, as permitted by 10 C.F.R. §§ 51.20(b)(14) and 51.22(b).

DISCUSSION: BCOC Contentions at 19-41; CP&L Contentions Response at 49-65; Staff Contentions Response at 16-20; Tr. at 153-70.

RULING: BCOC essentially agrees with the CP&L and Staff assertions that these contentions have been superseded by a Staff decision pursuant to 10 C.F.R. § 51.30 to issue an environmental assessment (EA) in the fall of this year. *See* Tr. at 153. We would agree because, in connection with such an assessment, the Staff will consider whether an EIS is needed relative to the CP&L amendment. *See* 10 C.F.R. § 51.31. CP&L and BCOC nonetheless do seek direction from the Board regarding two of the contentions. In CP&L's case, it seeks a dismissal with prejudice of EC-3, regarding the transfer of spent fuel from the Brunswick and Robinson facilities, asserting that consideration of the environmental impacts of storing fuel from these facilities was incorporated into the operating license proceeding for the Harris facility. *See* CP&L Contentions Response at 54, 57-59; *see also* Staff Contentions Response at 17. And for its part, BCOC seeks guidance on EC-5 regarding the Board's discretionary authority to order the Staff to prepare an EIS. *See* Tr. at 155.

In both instances, we decline the invitation to delve further into these contentions. Whatever validity these arguments may have in the context of further late-filed contentions submitted after the Staff's EA, *see* 10 C.F.R. § 2.714(b)(2)(iii), for now we consider any Board rulings to be premature. Accordingly, we dismiss all BCOC's contentions, but without prejudice to their being raised before the Board at some later juncture, as appropriate.

III. ADMINISTRATIVE MATTERS

As we noted during the prehearing conference, *see* Tr. at 171, this spent fuel capacity expansion proceeding is subject to the hybrid hearing process outlined in 10 C.F.R. Part 2, Subpart K, to the degree that any party wishes to invoke those procedures. Under Subpart K, following a 90-day discovery period, which can be extended upon a showing of exceptional circumstances, the parties simultaneously submit a detailed written summary of all facts, data, and arguments that each party intends to rely upon to support or refute the existence of a genuine and substantial dispute of fact regarding any admitted contentions. *See* 10 C.F.R. §§ 2.1111, 2.1113(a). Then, an oral argument is conducted by the presiding officer in which the parties address the question whether any of the issues require resolution in an adjudicatory proceeding because there are specific facts in genuine and substantial

dispute that can be resolved with sufficient accuracy only by the introduction of evidence. *See id.* § 2.1115(b). Thereafter, the presiding officer issues a decision that designates the disputed issues of fact for an evidentiary hearing and resolves any other issues. *See id.* § 2.1115(a).

Subpart K specifies that within 10 days of an order granting a hearing request in a proceeding such as this one, a party may invoke its procedures by filing a written request for an oral argument. *See id.* § 2.1109(a)(1). Accordingly, if CP&L, the Staff, or BCOC wishes to use the Subpart K procedures, it must file a request within 10 days of the date of this Memorandum and Order, or on or before *Thursday, July 22, 1999.*

IV. CONCLUSION

As a local governmental entity with a sovereign interest in protecting the health and welfare of its citizens and the environment within its boundaries, which come within approximately 17 miles of the Harris facility, Petitioner BCOC has made a showing sufficient to establish its standing to intervene as of right in this spent fuel pool expansion proceeding. Further, we find two of its eight contentions, TC-2 and TC-3, are supported by bases adequate to warrant further inquiry so as to be admitted for litigation in this proceeding. Accordingly, we grant BCOC's intervention petition and admit it as a party to this proceeding.

For the foregoing reasons, it is, this 12th day of July 1999, ORDERED that:

1. Relative to the contentions specified in paragraph two below, BCOC's hearing request/intervention petition is *granted* and BCOC is admitted as a party to this proceeding.

2. The following BCOC contentions are *admitted* for litigation in this proceeding: TC-2 and TC-3.

3. The following BCOC contentions are *rejected* as inadmissible for litigation in this proceeding: TC-1, EC-1, EC-2, EC-3, EC-4, and EC-5.

4. The parties are to file any request for an oral argument under 10 C.F.R. § 2.1109(a)(1) in accordance with the schedule established in Section III above.

5. In accordance with the provisions of 10 C.F.R. § 2.714a(a), as it rules upon an intervention petition, this Memorandum and Order may be appealed to the Commission within 10 days after it is served.

THE ATOMIC SAFETY
AND LICENSING BOARD[7]

G. Paul Bollwerk, III
ADMINISTRATIVE JUDGE

Frederick J. Shon
ADMINISTRATIVE JUDGE

Peter S. Lam
ADMINISTRATIVE JUDGE

Rockville, Maryland
July 12, 1999

[7] Copies of this Memorandum and Order were sent this date by Internet e-mail transmission to counsel for (1) Applicant CP&L; (2) Intervenor BCOC; and (3) the Staff.

Written data, views or comments for consideration by the committee may be submitted, preferably with 20 copies, to Joanne Goodell at the address provided below. Any such submissions received prior to the meeting will be provided to the members of the Committee and will be included in the record of the meeting. Because of the need to cover a wide variety of subjects in a period of time, there is usually insufficient time on the agenda for members of the public to address the committee orally. However, any such requests will be considered by the Chair who will determine whether or not time permits. Any request to make an oral presentation should state the amount of time desired, the capacity in which the person would appear, and a brief outline of the content of the presentation. Individuals with disabilities who need special accommodations should contact Theresa Berry (phone: 202–693–1999; FAX: 202–693–1641) one week before the meeting.

An official record of the meeting will be available for public inspection in the OSHA Technical Data Center (TDC) located in Room N2625 of the Department of the Labor Building (202–693–2350). For additional information contact: Joanne Goodell, Occupational Safety and Health Administration (OSHA); Room N–3641, 200 Constitution Avenue NW, Washington, D.C., 20210 (phone: 202–693–2400; FAX: 202–693–1641; e-mail joanne.goodell@osha-no.osha.gov; or at www.osha.gov).

Signed at Washington, D.C., this 7th day of January, 1999.

Charles N. Jeffress,

Assistant Secretary of Labor for Occupational Safety and Health.

[FR Doc. 99–744 Filed 1–12–99; 8:45 am]

BILLING CODE 4510–26–M

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice 99–013]

NASA Advisory Council, Minority Business Resource Advisory Committee; Meeting

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of meeting.

SUMMARY: In accordance with the Federal Advisory Committee Act, Pub. L. 92–463, as amended, the National Aeronautics and Space Administration announces a forthcoming meeting of the NASA Advisory Council, Minority Business Resource Advisory Committee.

DATES: Wednesday, January 27, 1999, 9:00 a.m. to 4:00 p.m. and Thursday, January 28, 1999, 9:00 a.m. to noon.

ADDRESSES: National Aeronautics and Space Administration, Lyndon B. Johnson Space Center, Building 1, Room 820, Houston, TX 77058–3696.

FOR FURTHER INFORMATION CONTACT: Mr. Ralph C. Thomas III, Code K, National Aeronautics and Space Administration, Washington, DC 20546, (202) 358–2088.

SUPPLEMENTARY INFORMATION: The meeting will be open to the public up to the seating capacity of the room. The agenda for the meeting is as follows:

—MBRAC Subpanel Reports
—Status of MBRAC Recommendations
—Special Issues
—Action Items
—Call to Order
—Reading of Minutes
—Agency Small Disadvantaged Business (SDB) Program
—Report of Chair
—Public Comment
—Center Directorate Reports
—Report on NASA FY 98 SDB Accomplishments

It is imperative that the meeting be held on these dates to accommodate the scheduling priorities of the key participants. Visitors will be requested to sign a visitors' register.

Dated: January 7, 1999.

Matthew M. Crouch,

Advisory Committee Management Officer, National Aeronautics and Space Administration.

[FR Doc. 99–741 Filed 1–12–99; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

[Notice 99–012]

Notice of Prospective Patent License

AGENCY: National Aeronautics and Space Administration.

ACTION: Notice of prospective patent license.

SUMMARY: NASA hereby gives notice that Benick Brands, Inc., of Glastonsbury, Connecticut, has applied for an exclusive license to practice the inventions described and claimed in U.S. Patent No. 5,772,912, entitled "Environmentally Friendly Anti-Icing Fluid," and in NASA Case No. ARC–12069–9GE, entitled "Anti-Icing Fluid or Deicing Fluid." Both inventions are assigned to the United States of America as represented by the Administrator of the National Aeronautics and Space Administration. Written objections tc the prospective grant should be sent t NASA Ames Research Center.

DATES: Responses to this notice shoul be received by March 15, 1999.

FOR FURTHER INFORMATION CONTACT: Kathleen Dal Bon, Patent Counsel, NASA Ames Research Center, Mail S 202A–3, Moffett Field, CA 94035–10(telephone (650) 604–5104.

Dated: January 7, 1999.

Edward A. Frankle,

General Counsel.

[FR Doc. 99–742 Filed 1–12–99; 8:45 am]

BILLING CODE 7510–01–P

NATIONAL BIPARTISAN COMMISSI(ON THE FUTURE OF MEDICARE

Public Meeting

The National Bipartisan Commissio on the Future of Medicare will hold a public meeting on Tuesday, January 2 1999 at the Cannon House Office Building, Cannon Caucus Room 340, Washington, DC. Please check the Commission's web site for additional information: *http://Medicare.Commission.Gov*

Tuesday, January 26, 1999, 9:00 a.m.

Tentative Agenda

Members of the Commission to discuss options to reform the Medicare program.

If you have any questions, please contact the Bipartisan Medicare Commission, ph: 202–252–3380.

I hereby authorize publication of the Medicare Commission meetings in the **Federal Register.**

Julie Hasler,

Office Manager, National Bipartisan Medic Commission.

[FR Doc. 99–681 Filed 1–12–99; 8:45 am]

BILLING CODE 1132–00–M

UNITED STATES NUCLEAR REGULATORY COMMISSION

[Docket No. 50–400]

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

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-63 issued to Carolina Power & Light (CP&L or the licensee) for operation of the Shearon Harris Nuclear Power Plan

located in Wake and Chatham Counties, North Carolina.

The proposed amendment would support a modification to the plant to increase the spent fuel storage capacity by adding rack modules to spent fuel pools (SFPs) "C" and "D" and placing the pools in service. In order to activate the pools, CP&L requests that the NRC review and approve the following:

i. Revised Technical Specification 5.6 to identify PWR burnup restrictions, 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.'

ii. 10 CFR 50.55a Alternative Plan to demonstrate acceptable level of quality and safety in the completion of the component cooling water (CCW) and SFP 'C' and 'D' cooling and cleanup system piping.

The cooling system for SFPs 'C' and 'D' cannot be N stamped in accordance with ASME Section III since some installation records are not available, a partial turnover was not performed when construction was halted following the cancellation of Unit 2 and CP&L's N certificate program was discontinued following completion of Unit 1.

iii. Unreviewed safety question for additional heat load on the CCW system. The acceptability of the 1.0 MBtu/hr heat load from SFPs 'C' and 'D' was demonstrated by the use of thermal-hydraulic analyses of the CCW system under various operating scenarios. The dynamic modeling used in the thermal-hydraulic analyses identified a decrease in the minimum required CCW system flow rate to the residual heat removal heat exchangers. This change has not been previously reviewed by the NRC and is deemed to constitute an unreviewed safety question.

Before issuance of the proposed license amendment, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

In the analysis of the safety issues concerning the expanded pool storage capacity within Harris' Fuel Handling Building, the following previously postulated accident scenarios have been considered:

a. A spent fuel assembly drop in a Spent Fuel Pool.

b. Loss of Spent Fuel Pool cooling flow.

c. A seismic event.

d. Misloaded fuel assembly.

The probability that any of the accidents in the above list can occur is not significantly increased by the activity itself. The probabilities of a seismic event or loss of Spent Fuel Pool cooling flow are not influenced by the proposed changes. The probabilities of accidental fuel assembly drops or misloadings are primarily influenced by the methods used to lift and move these loads. The method of handling loads during normal plant operations is not significantly changed, since the same equipment (i.e., Spent Fuel Handling Machine and tools) and procedures as those in current use in pools 'A' and 'B' will be used in pools 'C' and 'D'. Since the methods used to move loads during normal operations remain nearly the same as those used previously, there is no significant increase in the probability of an accident. Current shipping activities at the Harris Nuclear Plant will continue as previously licensed. The consequences of an accident involving shipping activities [are] not changed and there is no significant increase in the probability of an accident.

During rack installation, all work in the pool area will be controlled and performed in strict accordance with specific written procedures. Any movement of fuel assemblies which is required to be performed to support this activity (e.g., installation of racks) will be performed in the same manner as during normal refueling operations.

Accordingly, the proposed activity does not involve a significant increase in the probability of an accident previously evaluated.

The consequences of the previously postulated scenarios for an accidental drop of a fuel assembly in the Spent Fuel Pool have been re-evaluated for the proposed change. The results show that such the postulated accident of a fuel assembly striking the top of the storage racks will not distort the racks sufficiently to impair their functionality. The minimum subcriticality margin, K_{eff} less than or equal to 0.95, will be maintained. The structural damage to the Fuel Handling Building, pool liner, and fuel assembly resulting from a fuel assembly drop striking the pool floor or another assembly located within the racks is primarily dependent on the mass of the falling object and the drop height. Since these two parameters are not changed by the proposed activity from those considered previously, the structural damage to these items remains unchanged. The radiological dose at the exclusion area boundary will not be increased from those previously considered, since the pertinent fuel parameters remain unchanged. These dose levels remain "well within" the levels required by 10 CFR 100, paragraph 11, as defined in Section 15.7.4.II.1 of the Standard Review Plan. Thus, the results of the postulated fuel drop accidents remain acceptable and do not represent a significant increase in consequences from any of the same previously evaluated accidents that have been reviewed and found acceptable by the NRC.

The consequences of a loss of Spent Fuel Pool cooling have been evaluated and found to have no increase. The concern with this accident is a reduction of Spent Fuel Pool water inventory from bulk pool boiling resulting in uncovering fuel assemblies. This situation would lead to fuel failure and subsequent significant increase in offsite dose. Loss of spent fuel pool cooling at Harris is mitigated in the usual manner by ensuring that a sufficient time lapse exists between the loss of forced cooling and uncovering fuel. This period of time is compared against a reasonable period to re-establish cooling or supply an alternative water source. Evaluation of this accident usually includes determination of a time to boil, which in the case of pools 'C' and 'D' is in excess of 13 hours based on a consideration of end of plant life heat loads. This evaluation neglects any possible cooling from the connection to pools 'A' and 'B' through the transfer canal. The 13 hour period is much shorter than the onset of any significant increase in offsite dose, since once boiling begins it would have to continue unchecked until the pool surface was lowered to the point of exposing active fuel. The time to boil represents the onset of loss of pool water inventory and is commonly used as a gauge for establishing the comparison of consequences before and after a refueling project. The heatup rate in the Spent Fuel Pool is a nearly linear function of the fuel decay heat load. Subsequent to the proposed changes, the fuel decay heat load will increase because of the increase in the number assemblies from those considered from Pools 'A' and 'B' alone. The methodology used in the thermal-hydraulic analysis determined the maximum fuel decay heat loads. In the unlikely event that pool cooling is lost to pools 'C' and 'D', sufficient time will still be available for the operators to provide alternate means of cooling before the onset of pool boiling. Therefore, the proposed change represents no increase in the consequences of loss of pool cooling.

The consequences of a design basis seismic event are not increased. The consequences of this accident are evaluated on the basis of subsequent fuel damage or compromise of the fuel storage or building configurations leading to radiological or criticality concerns. The new racks have been analyzed in their new configuration and found safe during seismic motion. The fuel stored in these racks has been determined to remain intact and the racks maintain the fuel and fixed poison configurations subsequent to a seismic event. The structural capability of the pool and liner will not be exceeded under the appropriate combinations of dead weight, thermal, and seismic loads. The Fuel

Handling Building structure will remain intact during a seismic event and will continue to adequately support and protect the fuel racks, storage array, and pool moderator/coolant. Thus, the consequences of a seismic event are not increased.

Fuel misloading and mislocation accidents were previously credible occurrences, since fuel could be placed at an unintended storage location or could have been lowered outside and adjacent to a storage rack in Pools 'A' or 'B'. However, neither of these two scenarios previously represented any concern because of the flux trap style of the rack designs in these two pools. Similar procedures, equipment and methods of fuel movement will be used for Pools 'C' and 'D' as those used previously for Pools 'A' and 'B'. Therefore, the proposed activity does not represent any increase in the probability of occurrence. The proposed non-flux trap design racks for Pools 'C' and 'D' require administrative controls to ensure that fuel assemblies meet effective enrichment criteria prior to storage. Under these conditions, misloading of a fuel assembly by placement in an unintended storage cell has no significant consequences. Therefore, the only remaining potential mislocation of a fuel assembly is for an assembly to be lowered outside of and directly adjacent to a storage rack. This accident occurring in Pools 'C' or 'D' has been analyzed for the worst possible storage configuration subsequent to the proposed activity and it has been shown that the consequences remain acceptable with respect to the same criteria used previously. Thus, there is no increase in consequences for fuel mislocation or misloading.

Therefore it is concluded that the proposed changes do not significantly increase the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

To assess the possibility of new or different kind of accidents, a list of the important parameters required to ensure safe fuel storage was established. Safe fuel storage is defined here as providing an environment, which would not present any significant threats to workers or the general public (i.e., meeting the requirements of 10 CFR 100 and 10 CFR 20). Any new events, which would modify these parameters sufficiently to place them outside of the boundaries analyzed for normal conditions and/or outside of the boundaries previously considered for accidents would be considered to create the possibility of a new or different accident. The criticality and radiological safety evaluations were reviewed to establish the list of important parameters. The fuel configuration and the existence of the moderator/coolant were identified as the only two parameters, which were important to safe fuel storage. Significant modification of these two parameters represents the only possibility of an unsafe storage condition. Once the two important parameters were established, an additional step was taken to determine what events (which were not previously considered) could result in changes to the storage configuration or moderator/coolant presence during or subsequent to the proposed changes.

This process was adopted to ensure that the possibility of any new or different accident scenario or event would be identified. Due to the proposed activity, an accidental drop of a rack module during construction activity in the pool was considered as the only event which might represent a new or different kind of accident.

A construction accident resulting in a rack drop is an unlikely event. The proposed activity will utilize the defense-in-depth approach for these heavy loads. The defense-in-depth approach is intended to meet the requirements of NUREG–0612 and preclude the possibility of a rack drop. All movements of heavy loads over the pool will comply with the applicable administrative controls and guidelines (i.e. plant procedures, NUREG–0612, etc.). A temporary hoist and rack lifting rig will be introduced to lift and suspend the racks from the bridge of the Auxiliary Crane. These items have been designed in accordance with the requirements of NUREG–0612 and ANSI N14.6 and will be similar to those used recently to install storage rack modules in Pool 'B'.

The postulated rack drop event is commonly referred to as a "heavy load drop" over the pools. Heavy loads will not be allowed to travel over any racks containing fuel assemblies. The danger represented by this event is that the racks will drop to the pool floor and the pool structure will be compromised leading to loss of moderator/coolant, which is one of the two important parameters identified above. Although the analysis of this event has been performed and shown to be acceptable, the question of a new or different type of event is answered by determining whether heavy load drops over the pool have been considered previously. As stated above, heavy loads (storage rack modules) were recently installed in Pool 'B' using similar methods. Therefore, the rack drop does not represent a new or different kind of accident.

The proposed change does not alter the operating requirements of the plant or of the equipment credited in the mitigation of the design basis accidents. The proposed change does not affect any of the important parameters required to ensure safe fuel storage. Therefore, the potential for a new or previously unanalyzed accident is not created.

3. Involve a significant reduction in the margin of safety.

The function of the Spent Fuel Pool is to store the fuel assemblies in a subcritical and coolable configuration through all environmental and abnormal loadings, such as an earthquake or fuel assembly drop. The new rack design must meet all applicable requirements for safe storage and be functionally compatible with Pools 'C' and 'D'.

CP&L has Addressed the Safety Issues Related to the Expanded Pool Storage Capacity in the Following Areas:

1. Material, mechanical and structural considerations. The mechanical, material, and structural designs of the new racks have been reviewed in accordance with the applicable provisions of the NRC Guidance entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications" The rack materials used are compatible v the spent fuel assemblies and the Spent I Pool environment. The design of the new racks preserves the proper margin of safe during normal and abnormal loads. It has been shown that such loads will not invalidate the mechanical design and material selection to safely store fuel in a coolable and subcritical configuration.

2. Nuclear Criticality

The methodology used in the criticality analysis of the expanded Spent Fuel Pool meets the appropriate NRC guidelines an the ANSI standards (GDC 62, NUREG 08(Section 9.1.2, the OT Position for Review Acceptance of Spent Fuel Storage and Handling Applications, Reg. Guide 1.13, ANSI/ANS 8.17). The margin of safety for subcriticality is maintained by having the neutron multiplication factor equal to, or than, 0.95 under all accident conditions, including uncertainties. This criterion is same as that used previously to establish criticality safety evaluation acceptance an remains satisfied for all analyzed acciden

3. Thermal-hydraulic and Pool Cooling

The thermal-hydraulic and cooling evaluation of the pools demonstrated that pools can be maintained below the specif thermal limits under the conditions of the maximum heat load and during all credib accident sequences and seismic events. T pool temperature will not exceed 137°F during the highest heat load conditions. T maximum local water temperature in the channel will remain below the boiling poi The fuel will not undergo any significant heat up after an accidental drop of a fuel assembly on top of the rack blocking the f path. A loss of cooling to the pool will all sufficient time (>13 hours) for the operato to intervene and line up alternate cooling paths and the means of inventory make-up before the onset of pool boiling. The therm limits specified for the evaluations perforr to support the proposed activity are the sa as those that were used in the previous evaluations. It has also been demonstrated that adequate margin exists in the Unit 1 CCW system to support near term operatic of the pools subject to the requirements of proposed changes to the Technical Specifications.

Based on the preceding discussion it is concluded that this activity does not invol a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments receive within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By February 12, 1999, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

The Commission hereby provides such notice that this is a proceeding on an application for a license amendment falling within the scope of section 134 of the Nuclear Waste Policy Act of 1982 (NWPA), 42 U.S.C. 10154. Under section 134 of the NWPA, the

Commission, at the request of any party to the proceeding, must use hybrid hearing procedures with respect to "any matter which the Commission determines to be in controversy among the parties."

The hybrid procedures in section 134 provide for oral argument on matters in controversy, preceded by discovery under the Commission's rules and the designation, following argument of only those factual issues that involve a genuine and substantial dispute, together with any remaining questions of law, to be resolved in an adjudicatory hearing. Actual adjudicatory hearings are to be held on only those issues found to meet the criteria of section 134 and set for hearing after oral argument.

The Commission's rules implementing section 134 of the NWPA are found in 10 CFR Part 2, Subpart K, "Hybrid Hearing Procedures for Expansion of Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors" (published at 50 FR 41662 dated October 15, 1985). Under those rules, any party to the proceeding may invoke the hybrid hearing procedures by filing with the presiding officer a written request for oral argument under 10 CFR 2.1109. To be timely, the request must be filed within ten (10) days of an order granting a request for hearing or petition to intervene. The presiding officer must grant a timely request for oral argument. The presiding officer may grant an untimely request for oral argument only upon a showing of good cause by the requesting party for the failure to file on time and after providing the other parties an opportunity to respond to the untimely request. If the presiding officer grants a request for oral argument, any hearing held on the application must be conducted in accordance with the hybrid hearing procedures. In essence, those procedures limit the time available for discovery and require that an oral argument be held to determine whether any contentions must be resolved in an adjudicatory hearing. If no party to the proceeding timely requests oral argument, and if all untimely requests for oral argument are denied, then the usual procedures in 10 CFR Part 2, Subpart G apply.

For further details with respect to this action, see the application for amendment dated December 23, 1998, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Dated at Rockville, Maryland, this 7th day of January 1999.

For the Nuclear Regulatory Commission.

Scott Flanders,

Project Manager, Project Directorate II-3, Division of Reactor Projects—I/II, Office of Nuclear Reactor Regulation.

[FR Doc. 99-758 Filed 1-12-99; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

[Docket Number 40-8102]

Exxon Coal and Minerals Company

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of receipt of Exxon Coal and Minerals Company's application for establishing alternate concentration limits in source material license SUA-1139 for the Highland Uranium Mill in Converse County, Wyoming; notice of opportunity for a hearing.

SUMMARY: Notice is hereby given that the U.S. Nuclear Regulatory Commission (NRC) has received, by letter dated December 18, 1998, an application from Exxon Coal and Minerals Company (ECMC) to establish Alternate Concentration Limits (ACLs) for nickel, radium (Ra 226+228), and natural uranium (UNAT); and amend accordingly Source Material License No. SUA-1139 for the Highland uranium mill.

FOR FURTHER INFORMATION CONTACT: Mohammad W. Haque, Uranium Recovery Branch, Division of Waste Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415-6640.

SUPPLEMENTARY INFORMATION: ECMC's application to amend Source Material License SUA-1139, which describes the proposed change and the reasons for the request, is being made available for public inspection at NRC's Public Document Room at 2120 L Street, N.W. (Lower Level), Washington, DC 20555.

The NRC hereby provides notice of an opportunity for a hearing on the license amendment under the provisions of 10 CFR Part 2, Subpart L, "Informal Hearing Procedures for Adjudications in Materials and Operator Licensing Proceedings." Pursuant to § 2.1205(a), any person whose interest may be affected by this proceeding may file a request for a hearing. In accordance with § 2.1205(c), a request for hearing must be filed within 30 days of the publication of this notice in the Federal Register. The request for a hearing must be filed with the Office of the Secretary, either:

(1) By delivery to the Docketing ar Service Branch of the Office of the Secretary at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; or

(2) By mail or telegram addressed the Secretary, U.S. Nuclear Regulato Commission, Washington, DC 20555 0001, Attention: Docketing and Serv: Branch.

In accordance with 10 CFR 2.1205 each request for a hearing must also l served, by delivering it personally, o mail, to:

(1) The applicant, Exxon Coal and Minerals Company, P.O. Box 1314, Houston, Texas 77251-1314, Attenti David Range; and

(2) The NRC staff, by delivery to th Executive Director for Operations, O White Flint North, 11555 Rockville Pike, Rockville, MD 20852, or by mai addressed to the Executive Director f Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

In addition to meeting other applicable requirements of 10 CFR Pa 2 of NRC's regulations, a request for a hearing filed by a person other than a applicant must describe in detail:

(1) The interest of the requestor in t proceeding;

(2) How that interest may be affecte by the results of the proceeding, including the reasons why the reques should be permitted a hearing, with particular reference to the factors set in § 2.1205(g);

(3) The requestor's areas of concern about the licensing activity that is the subject matter of the proceeding; and

(4) The circumstances establishing that the request for a hearing is timely in accordance with § 2.1205(c).

The request must also set forth the specific aspect or aspects of the subje matter of the proceeding as to which petitioner wishes a hearing.

Dated at Rockville, Maryland, this 5th d of January 1999.

N. King Stablein,

Acting Chief, Uranium Recovery Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 99-756 Filed 1-12-99; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

[Docket No. 72-09]

Public Service Company of Colorado Fort St. Vrain Independent Spent Fue Storage Installation; Exemption

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Public Service Company of Colorad (PSCo, the licensee) holds Materials

March 16, 2001

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

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

ORANGE COUNTY'S PETITION FOR REVIEW OF LBP-00-12, LBP-00-19, AND LBP-01-09

Introduction

Pursuant to 10 C.F.R. § 2.786 (b)(1), the Board of Commissioners of Orange County, North Carolina ("Orange County") petitions the Nuclear Regulatory Commission ("NRC" or "Commission") for review of three Licensing Board decisions in this proceeding: LBP-00-12, Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing), 51 NRC 247 (2000); LBP-00-19 (Ruling on Late-Filed Environmental Contentions), 52 NRC 85 (2000); and LBP-01-09, Memorandum and Order (Denying Request for Evidentiary Hearing and Terminating Proceeding) (March 1, 2001).

I. FACTUAL AND PROCEDURAL BACKGROUND

A. Factual Background

In this license amendment proceeding, Carolina Power & Light Company ("CP&L"), seeks to activate two spent fuel pools (labeled "C" and "D") for which it abandoned its construction permit application and quality assurance program in the early 1980's. The piping and equipment for pools C and D have sat idle since CP&L abandoned construction of Units 2, 3, and 4 in the early 1980's.

Pools A and B now have a combined capacity of 1,128 PWR spent fuel

assemblies and 2,541 BWR assemblies. The proposed license amendment would allow CP&L to use pools C and D for storage of an additional 1,952 PWR spent fuel assemblies and 2,763 BWR assemblies. This would bring the amount of fuel to be stored at Harris to 8,343 assemblies, which is over a thousand more assemblies than were assumed in the 1983 FEIS.[1] In order to activate pools C and D, CP&L must complete construction of the cooling system for the pools.

B. Procedural Background

The NRC Staff noticed the proposed license amendment on January 13, 1999, at 64 Fed. Reg. 2,237. Orange County filed a request for a hearing on the proposed license amendment, which was granted with respect to criticality prevention and quality assurance issues in LBP-99-25, 50 NRC 25 (1999).[2] Following a Subpart K proceeding, the Licensing Board dismissed both technical contentions in LBP-00-12.[3]

On December 15, 1999, the NRC Staff issued an Environmental Assessment ("EA") and Finding of No Significant Impact ("FONSI") for the proposed license amendment, which concluded that the proposed expansion of spent fuel storage capacity

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1 *See* CP&L License Amendment Application, Enclosure 1 at 3 (December 23, 1998). Pool D will not be filled until a later "campaign," by which time CP&L will also need to have obtained a license amendment permitting it to exceed the license's current 1.0 million BTU/hour limit on the heat load in pools C and D. At that point, however, no further licensing action will be needed regarding the number of spent fuel assemblies that can be stored in either pool C or D. The number of spent fuel assemblies permitted to be stored at the Harris site will have been previously approved in this license amendment proceeding.

2 The Board admitted Contentions TC-2 and TC-3. Contention TC-2 asserted, *inter alia*, that CP&L's reliance on control of burnup levels for criticality prevention violates GDC 62, because it constitutes an administrative measure and is therefore prohibited by GDC 62. Contention TC-3 asserted, *inter alia*, that CP&L's license amendment application does not comply with Appendix B 10 C.F.R. Part 50, because CP&L has not maintained piping and equipment in conformance with lay-up requirements of Criteria XIII, XVI, and XVII. The County also filed several environmental contentions, which were dismissed as premature. *Id.*

3 51 NRC 247 (2000). On May 22, 2000, Orange County filed a petition for review. The petition was denied without prejudice, on grounds of prematurity. *See* CLI-00-11, 51 NRC 297 (2000). Later, Orange County was given the opportunity to participate as an amicus on appeal of LBP-00-26, a decision interpreting criticality prevention requirements in a spent fuel pool expansion case involving the Millstone nuclear power plant. *See Northeast Nuclear Energy Company* (Millstone Nuclear Power Station, Unit 3),

at Harris "will not significantly increase the probability or consequences of accidents.[4]

Orange County subsequently submitted several contentions challenging the adequacy of the EA.[5] On August 7, 2000, in LBP-00-19, the Licensing Board admitted Contention EC-1 (renumbered Contention EC-6). Contention EC-6 charged that the EA failed to take into consideration new information and changed circumstances, showing the foreseeable potential for a severe spent fuel pool accident following a degraded core accident with containment failure or bypass.[6] The Board found that Orange County had established "an adequate basis to allow merits litigation on whether the following accident sequence is not 'remote and speculative' so that a further environmental analysis of the CP&L pool expansion amendment is required," with respect to the following seven-step accident scenario:

> 1) a degraded core accident;
> 2) containment failure or bypass;
> 3) loss of all spent fuel cooling and makeup systems;
> 4) extreme radiation doses precluding personnel access;
> 5) inability to restart any pool cooling or makeup systems due to extreme radiation doses;
> 6) loss of most or all pool water through evaporation; and
> 7) initiation of an exothermic oxidation reaction in pools C and D.

52 NRC at 95. The Board invoked the summary procedures of Subpart K to 10 C.F.R. Part 2, and required the parties to file written presentations and deliver oral argument to determine whether the hearing should go forward. In support of its position, Orange County filed an extensive legal brief and detailed expert report by Dr. Gordon

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CLI-01-03 (January 17, 2001). Orange County filed an amicus brief on February 7, 2001.

4 Environmental Assessment Related to Expanding the Spent Fuel Pool Storage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 6.

5 Orange County's Request for Admission of Late-Filed Environmental Contentions (January 31, 2000) ("Environmental Contentions"). These contentions were also supported by an expert report and declaration prepared by Dr. Thompson.

6 LBP-00-19, Memorandum and Order (Ruling on Late-Filed Environmental Contentions), 52 NRC 85 (2000).

Thompson.[7] Dr. Thompson's report relied to a significant extent on information and analyses previously prepared by CP&L and the NRC Staff. His report presented substantial and material evidence that the probability of an exothermic reaction in the spent fuel pools, leading to a massive release of radiation from the pools, is foreseeable, and may not be disregarded as a remote and speculative event. Orange County's legal brief also showed that the NRC Staff unlawfully relied on assumptions regarding doses to workers during accidents that are inconsistent with the requirements of the National Environmental Policy Act ("NEPA"). *See* Orange County's Summary at 31-38.

The NRC Staff and CP&L also filed legal and evidentiary presentations, arguing that the probability of a severe spent fuel pool accident is too small to warrant consideration. Although there were some areas of agreement between the parties, their analyses showed stark differences in the information relied on, analytical approach, and results reached. At the oral argument on December 7, 2000, Orange County pointed out that CP&L's and the Staff's technical analyses contained considerable omissions and deficiencies, including failure to provide transparent technical analysis or actual calculations regarding some parameters, and oversimplification of accident behavior.

On March 1, 2001, the Licensing Board issued LBP-01-09. The decision went through each of the seven accident steps the parties had been asked to address, and compared the evidence presented by the three parties. For each of the seven steps, the

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7 *See* Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with Respect to the Need to Prepare an Environmental Impact Statement to Address the Increased Risk of a Spent Fuel Pool Accident (November 20, 2000) ("Orange County's Summary re: Contention EC-6"); BCOC Exhibit 1, Declaration of Dr. Gordon Thompson (November 20, 2000) ("Thompson Declaration"); BCOC Exhibit 2, G. Thompson, *The Potential for a Large, Atmospheric Release of Radioactive Material From Spent Fuel Pools at the Harris Nuclear Power Plant: The Case of a Pool Release initiated by a Severe Reactor Accident* (November 20, 2000) ("Thompson Report").

Board ruled that Orange County had not met the NRC's standard for proceeding to an evidentiary hearing. The Board accepted the NRC Staff's calculation that the probability of the seven-step accident is "conservatively in the range of" 2.0E-7 per reactor year, and found that this level of probability falls within the realm of "remote and speculative" events not cognizable under NEPA. LBP-01-09, slip op. at 34-36.

II. THE COMMISSION SHOULD GRANT REVIEW.

The three decisions below meet the Commission's standard for taking discretionary review in 10 C.F.R. § 2.786(b)(4), because they raise substantial questions with respect to their reliance on legal errors and clear factual errors. They also raise substantial and important questions of law, discretion and policy.

A. LBP-00-12 Raises Substantial Questions of Legal Error And Clearly Erroneous Factual Error.

The Licensing Board made several significant legal errors in LBP-00-12. First, the Board ruled that General Design Criterion 62 permits the administrative and procedural measures on which CP&L proposes to rely for criticality prevention. The Board's interpretation of GDC 62 is contrary to the plain language and regulatory history of GDC 62, which restricts criticality prevention measures to "physical systems and processes." CP&L's proposed method of burnup/enrichment control constitutes a non-physical and procedural criticality prevention measure that the Commission intended to exclude from the scope of GDC 62. As discussed in note 3, *supra,* the Commission has already demonstrated that it considers this issue worthy of review, by taking review in the Millstone license amendment case. For the same reasons, it should take review here.

Second, the Board clearly erred by ignoring a significant portion of Orange County's evidentiary case on quality assurance issues, *i.e.*, that video-camera inspections

conducted by CP&L were deficient because they covered only the embedded welds, and did not examine embedded piping. The Board unquestioningly assumed that piping was inspected along with the welds, without addressing any of the County's considerable evidence that only the welds were inspected.[8]

Third, the Board erred in refusing to consider the County's argument that CP&L must seek a construction permit amendment in order to use piping and equipment that were abandoned in the early 1980's.[9]

B. LBP-00-19 Raises Substantial Questions of Legal Error.

In Contention EC-6, Orange County challenged the NRC Staff's refusal to prepare an Environmental Impact Statement ("EIS") for the proposed license amendment, on the ground of new information and changed circumstances showing that the probability of a spent fuel pool accident is higher than previously considered by the Staff, and is not remote and speculative. *See* Environmental Contentions at 1-16. As an illustration for its assertion, Orange County posited a seven-step accident scenario involving a degraded core reactor accident with containment failure or bypass. Orange County asserted that such an accident would establish a "lower bound of probability" of a severe pool accident, because such an accident would almost certainly lead to a pool accident. Environmental Contentions at 11. The contention also referred to other causes of spent

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8 See Orange County's Summary Etc. Regarding Quality Assurance Issues at 36-44 (January 4, 2000). The Board also completely failed to address the County's evidence that CP&L failed to follow its own weld inspection procedures, and that a single recent water test is insufficient to demonstrate that the pipes were free of corrosive agents during a 15-year period when no records were kept. *See* Orange County's QA Summary at 29-35, 45-51. In addition, the Board's dismissal of potential leakage from spent fuel cooling pipes as insignificant constitutes an unlawful and unacceptable relaxing of NRC quality assurance standards.

9 According to the Board, this argument amounted to a new contention for which the County failed to address the late-filing standard. To the contrary, the argument constituted a permissible response to CP&L's and the Staff's assertions that CP&L's abandonment of its construction permit and QA program is of "no consequence" to this licensing proceeding. The Board's ruling erroneously sidestepped this critical issue by characterizing it as a late-filed contention.

fuel pool accidents, and NRC studies which evaluated them. *Id.* at 9-11.

In LBP-00-19, the Licensing Board admitted Contention EC-6, but only for the purpose of addressing the probability of the illustrative seven-step accident scenario identified by Orange County in the contention. 52 NRC at 97-98. The Board did not admit the broader issue of the overall probability of a spent fuel pool accident at Harris, even though Orange County had pled the issue with basis and specificity.[10] By restricting the scope of admitted Contention EC-6 to the probability of a single accident scenario, the Board arbitrarily excluded consideration of the contribution of other accident scenarios to the overall probability of a pool fire accident at Harris.[11]

The Board's ruling is based on an overly narrow reading of *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990); and *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129 (1990). While it may be reasonable to require an intervenor to support a contention challenging existing NRC probability calculations with a plausible accident scenario, it does not follow from those decisions that the contribution of other, uncontested probability calculations to the overall probability of a spent fuel pool fire is beyond the scope of any contention.

C. LBP-01-09 Raises Substantial Questions of Legal Error and Clear Factual Error.

In LBP-01-09, the Licensing Board committed both legal error and clear factual

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10 Consistent with its decision in LBP-00-19, in LBP-01-09, the Board addressed only the probability of the seven-step scenario admitted in LBP-00-19. The Board made no attempt to address the overall probability of a severe pool accident. Nor did it address Orange County's proposition in Contention EC-6 that the question of whether a degraded-core accident with containment bypass constituted a lower bound of probability of a pool accident. This is an important omission, because, as the Board acknowledges, the Staff has made probability estimates for other spent fuel pool accident scenarios that are higher than its estimate for the seven-step accident scenario admitted in LBP-00-19. *See* LBP-01-09, slip op. at 39.

11 For instance, NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82: Beyond

error, in three major respects. First, the Board misapplied the standard applied in Subpart K proceedings for determining whether to order a hearing. Under 10 C.F.R. § 2.1115, the Board was required to review written evidentiary and legal presentations and conduct an oral argument to determine whether there is a genuine and substantial dispute of material fact between the parties, that can only be resolved in a hearing. The purpose of the proceeding is to separate "genuine factual issues (for subsequent adjudication) from issues of policy or law and/or frivolous factual issues."[12] Although the intervenor bears the burden of showing a genuine and substantial material issue of fact that should go to a hearing, the license amendment applicant bears the ultimate burden of proof, and the Staff also bears the burden of proof on NEPA issues. LBP-01-09, slip op. at 10-12. The Commission has likened the Subpart K process to summary disposition.[13]

Here, the Licensing Board went far beyond the bounds of determining whether there is a genuine and substantial dispute of material fact. Instead, the Board entered the merits of the dispute, weighed the credibility of each side in the dispute, and chose for one of the parties.[14] The Board had no basis for making this choice, other than its own predilections. Resolution of disputed factual issues must be reserved for trial, after hearing testimony from the experts.[15] Orange County met its burden of demonstrating a genuine and substantial dispute of material fact that could only be resolved at a full

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Design Basis Accidents in Spent Fuel Pools (1989), calculates the probability of a spent fuel pool accident at 2E-06 per reactor year. *See* LBP-01-09, slip op. at 39.

12 Proposed Rule, Hybrid Hearing Procedures for Expansion of Onsite Spent Fuel Storage Capacity at Civilian Nuclear Power Plants, 48 Fed. Reg. 54,499, 54,451 (December 5, 1983).

13 Final Rule, Hybrid Hearing Procedures for Expansion of Onsite Spent Fuel Storage Capacity at Civilian Nuclear Power Plants, 50 Fed. Reg. 41,662, 41,669 (October 15, 1985).

14 For example, the Board consistently uses terminology reflecting a weighing of the evidence, rather than the identification of genuine and substantial issues of fact. Its determinations are based on the basis of the comparative credibility (pages 21, 26, 31); reasonableness (pages 23, 26, 33), and persuasiveness (page 23) of the parties positions. The Board also makes judgments about the comparative complexity of the parties' analyses. *Id.*, slip op. at 26.

15 *Sequoyah Fuels Corp. and General Atomics* (Gore, Oklahoma Site Decontamination and

evidentiary hearing. By reaching the merits of the dispute, the Licensing Board illegally shifted the ultimate burden of proof from CP&L and the NRC Staff to Orange County.

Second, even if it were appropriate for the Board to go to the merits of the dispute, the Board's ruling was based on an arbitrary and capricious selection of facts favorable to the NRC Staff.[16] In all areas where there was a dispute between the parties as to the best probability estimate for a given step in the seven-step scenario, the Licensing Board ignored or summarily rejected Orange County's factual evidence, without providing a reasoned explanation.[17] This lack of accountability flagrantly violates NRC precedent.[18]

Decommissioning Funding), LBP-94-17, 39 NRC 359, 361 (1994).

16 The Board makes a point of not relying on CP&L's analysis. *See* LBP-01-09, slip op. at 16-17. However, throughout the decision, it summarizes CP&L's analysis uncritically, thus giving the strong impression that CP&L's analysis would constitute independent and additional grounds to rule against a hearing. This message has no basis in fact. As discussed during oral argument, CP&L's analysis is fatally defective because it omits critical data and calculations that are necessary in order to make a reasonable evaluation of the reliability of its conclusions. *See* Transcript of December 7, 2000, oral argument at 468-76, 483-83, 596, 493-95.

17 For example, the Board completely ignored Orange County's evidence regarding the type of analysis needed to make a credible probability estimate for a spent fuel pool fire. *See* Thompson 2000 Report, Section 3. The Board also misrepresents Orange County's position on the potential occurrence of a degraded-core accident at Harris. At pages 17-19, the Board misrepresents the County's estimate of the overall probability of four *selected* sequences as the County's estimate of the probability of core degradation through *all possible* sequences. The Board also misrepresents Orange County's position about the loss of spent fuel pool cooling that accompanies the four selected sequences. In discussing these sequences, the ASLB states that all of them "lead finally to a loss of cooling to the fuel pools." To the contrary, each sequence involves a loss of cooling to the fuel pools from the beginning of the accident sequence until the occurrence of core degradation and potentially beyond. This point is significant because it bears upon the potential for recover of pool cooling, and thereby on the probability of a pool fire.

At pages 19-21, the ASLB shows a complete misunderstanding of Orange County's position regarding the potential for a degraded-core accident to lead to containment bypass. The Board characterizes Orange County's analysis as "too simplistic," but fails to address the fact that Orange County drew its analysis from an NRC Staff analysis. It also fails to address the fact that the NRC Staff did not address the significance of its own analysis in its evidentiary presentation. In footnote 5, the Board again shows a poor understanding of the literature regarding the effect of high levels of fuel burnup on the release of radioactive material from a reactor to the atmosphere during a degraded-core accident. The Board ignored a relevant study, NUREG-1465, and misinterpreted a research paper on the subject. As a result, the Board arbitrarily resolved a genuine and substantial factual dispute between the parties based on its own arbitrary and ill-informed weighing of the merits of the evidence.

At pages 26-27, the Licensing Board credited the NRC Staff's analysis of deposition patterns of radioactive material onsite, partly on the ground that Orange County had not itself modeled the deposition patterns. The Board completely disregarded Orange County's criticisms of the method used by the NRC Staff to estimate onsite radiation levels. Instead, it shifted to Orange County the burden of proving that radiation levels would be higher than calculated by the Staff. These are but some examples of the many factual

Finally, the Board based its decision on a critical assumption that is inconsistent with NEPA. In order to come up with a very low probability calculation for a spent fuel pool fire, the NRC Staff assumed the workers would incur doses above regulatory limits in order to stop the accident from progressing to that point.19 *See* LBP-01-09, slip op. at 28-30. In approving a probability calculation that is based on this assumption, the Licensing Board unlawfully accepted one type of environmental harm (radiation exposure to plant workers beyond regulatory limits) as the justification for avoiding another type of environmental harm (harm to the general public and the environment caused by radiological releases from the spent fuel pools), without going through the process of fully disclosing these competing harms in an EIS.

III. CONCLUSION

Not only do the Board's decisions in this proceeding violate the law, but they raise significant concerns about the Commission's practice and policy for addressing the risks of high-density spent fuel pool storage. Accordingly, for the foregoing reasons, the Commission should take review of LBP-01-09.

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errors made by the Licensing Board. It would be impossible, given the page limitations of this petition, to list them all. These examples illustrate the extreme arbitrariness of the Licensing Board's handling of a complex and fact-intensive dispute between the parties to this proceeding.

18 *Public Service Electric and Gas Company, Atlantic City Electric Company* (Hope Creek Generating Station, Units 1 and 2), ALAB-429, 6 NRC 229, 237 (1977) ("a Licensing Board must do more than reach conclusions; it must 'confront the facts.'"") (citation omitted). As in *Hope Creek*, this record "is devoid of any systematic analysis by expert witnesses for either the applicant or the staff of the differences between [two studies at issue in the hearing]." *Id.* Instead, without the benefit of the substantial expertise required to evaluate the issues at hand, the Board makes its own arbitrary judgments on the merits of the case.

19 A dose of 5 rems TEDE per year per year is the occupational dose limit established by NRC standards for protection of worker safety and health. *See* 10 C.F.R. § 20.1201(a)(1)(i). The NRC Staff assumes that workers will incur a dose of 25 rems in an accident at Harris. The issue here is not whether workers would be willing to incur such doses during a real accident, but whether such high doses can be assumed *for purposes of avoiding the preparation of an EIS.*

Respectfully submitted,

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March 16, 2001

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

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

NRC STAFF BRIEF IN RESPONSE TO COMMISSION ORDER OF FEBRUARY 14, 2001

Susan L. Uttal
Counsel for NRC Staff

February 28, 2001

[Cover only]

December 22, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSION

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

ORANGE COUNTY'S PETITION FOR REVIEW AND REQUEST FOR IMMEDIATE SUSPENSION AND STAY OF THE NRC STAFF'S NO SIGNIFICANT HAZARDS DETERMINATION AND ISSUANCE OF LICENSE AMENDMENT FOR HARRIS SPENT FUEL POOL EXPANSION

Introduction and Summary

Pursuant to 10 C.F.R. § 50.58(b)(6) and the Commission's inherent supervisory authority, the Board of Commissioners of Orange County, North Carolina ("Orange County") hereby petitions the Nuclear Regulatory Commission ("NRC" or "Commission") for review of the NRC Staff's No Significant Hazards Determination ("NSH Determination") and issuance of a license amendment for the expansion of spent fuel pool storage capacity at the Shearon Harris nuclear power plant.[1] The Commission should take review of the NSH Determination because it fails to satisfy the NRC's criteria in 10 C.F.R. § 50.92, and because it clearly violates the National Environmental Policy Act ("NEPA").

Review is also warranted because the Staff's decision violates all notions of agency

1 United States Nuclear Regulatory Commission, Carolina Power & Light Company, Docket No. 50-400, Notice of Issuance of Amendment to Facility Operating License and Final Determination of No Significant Hazards Consideration (December 21, 2000). A discussion of the Staff's No Significant Hazards analysis is also found in Section 4.0 of the Staff's Safety Analysis Report ("SER"), which is attached to a letter from Richard J. Laufer, NRC Project Manager, to James Scarola, CP&L Vice President (December 21, 2000).

proceeding in which the central contested issue is also determinative of the validity of the NSH determination, the Staff undermines the integrity of the adjudicatory proceeding to which Orange County has in good faith submitted and devoted its time and resources. The Staff's affront to the legal process is all the more egregious in light of the procedural status of this proceeding: the parties have completed discovery, filed extensive Subpart K presentations, and participated in a day-long oral argument; and a decision from the Licensing Board on the Subpart K submissions is imminent within a matter of weeks. If allowed to stand, the NSH Determination would make a complete mockery of the Commission's hearing process.

Orange County also meets the Commission's standard for a stay or suspension of the license amendment, pending the outcome of the adjudicatory proceeding. The County's request for suspension of the license amendment is supported by the declaration of Dr. Gordon Thompson, a nuclear safety expert with extensive experience in the field of probabilistic risk assessment.[2] Dr. Thompson's Declaration demonstrates that the proposed license amendment would cause irreparable injury by significantly increasing the consequences of a severe spent fuel pool accident.

I. FACTUAL AND PROCEDURAL BACKGROUND

A. Factual Background

In this license amendment proceeding, Carolina Power & Light Company ("CP&L"), seeks to activate two spent fuel pools (labeled "C" and "D") for which it abandoned its construction permit application and quality assurance program in the early 1980's. Pools A and

2 Declaration of 22 December 2000 by Dr. Gordon Thompson Regarding the Potential for a Severe Accident at Spent Fuel Pools C and D At the Harris Nuclear Power Plant (December 22, 2000) ("Thompson December 22 Declaration").

B now have a combined capacity of 1,128 PWR spent fuel assemblies and 2,541 BWR assemblies. The proposed license amendment would allow CP&L to use pools C and D for storage of an additional 1,952 PWR spent fuel assemblies and 2,763 BWR assemblies. This would bring the amount of fuel to be stored at Harris to 8,343 assemblies, which is over a thousand more assemblies than were assumed in the 1983 FEIS.[3]

B. Procedural Background

The NRC Staff noticed the proposed license amendment on January 13, 1999, at 64 Fed. Reg. 2,237. The notice was accompanied by a proposed No Significant Hazards Determination. Orange County submitted detailed comments on the proposed No Significant Hazards Determination, arguing that the proposed license amendment did not satisfy the NRC's regulatory criteria in 10 C.F.R. § 50.92(c).[4] Orange County's comments were supported by a declaration prepared by Dr. Thompson, which provided detailed discussion of the technical bases for his view that the proposed license amendment raises significant hazard considerations.[5]

Orange County filed a request for a hearing on the proposed license amendment, which was granted with respect to criticality prevention and quality assurance issues in LBP-99-25, 50

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3 *See* CP&L License Amendment Application, Enclosure 1 at 3 (December 23, 1998). Pool D will not be filled until a later "campaign," by which time CP&L will also need to have obtained a license amendment permitting it to exceed the license's current 1.0 million BTU/hour limit on the heat load in pools C and D. At that point, however, no further licensing action will be needed regarding the number of spent fuel assemblies that can be stored in either pool C or D. The number of spent fuel assemblies permitted to be stored at the Harris site will have been previously approved in this license amendment proceeding.

4 *See* Orange County's Comments in Opposition to No Significant Hazards Determination and Conditional Request for Stay of Effectiveness (February 12, 1999).

5 Declaration of Dr. Gordon Thompson (February 12, 1999) ("Thompson NSH Declaration").

NRC 25 (1999).[6] Following a Subpart K proceeding, the Licensing Board dismissed both technical contentions. LBP-00-12, 51 NRC 247 (2000). Orange County filed a petition for review, which was denied on grounds of prematurity.[7]

On December 15, 1999, the NRC Staff issued an Environmental Assessment ("EA") and Finding of No Significant Impact ("FONSI") for the proposed license amendment.[8] In the EA, the NRC Staff concluded that the proposed expansion of spent fuel storage capacity at the Shearon Harris nuclear power plant:

> will not significantly increase the probability or consequences of accidents, no changes are being made in the types of any effluents that may be released offsite, and there is no significant increase in occupational or public radiation exposure.

Id. at 6.

Orange County subsequently submitted several contentions challenging the adequacy of the EA.[9] On August 7, 2000, in LBP-00-19, the Licensing Board admitted Contention EC-6, which charged that the EA failed to take into consideration new information and changed circumstances, showing the potential for a severe spent fuel pool accident following a core melt

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6 The Board admitted Contentions TC-2 and TC-3. Contention TC-2 asserted, *inter alia*, that CP&L's reliance on control of burnup levels for criticality prevention violates GDC 62, because it constitutes an administrative measure and is therefore prohibited by GDC 62. Contention TC-3 asserted, *inter alia,* that CP&L's license amendment application does not comply with Appendix B 10 C.F.R. Part 50, because CP&L has not maintained piping and equipment in conformance with lay-up requirements of Criteria XIII, XVI, and XVII. The County also filed several environmental contentions, which were dismissed as premature. *Id.*

7 Orange County's Petition for Review of LBP-00-12 (May 22, 2000).

8 Environmental Assessment Related to Expanding the Spent Fuel Pool Storage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432).

9 Orange County's Request for Admission of Late-Filed Environmental Contentions (January 31, 2000). These contentions were also supported by an expert report and declaration prepared by Dr. Thompson.

accident with containment bypass.[10] The Board found that Orange County had established "an adequate basis to allow merits litigation on whether the following accident sequence is not 'remote and speculative' so that a further environmental analysis of the CP&L pool expansion amendment is required," with respect to the following accident scenario:

> 1) a degraded core accident;
> 2) containment failure or bypass;
> 3) loss of all spent fuel cooling and makeup systems;
> 4) extreme radiation doses precluding personnel access;
> 5) inability to restart any pool cooling or makeup systems due to extreme radiation doses;
> 6) loss of most or all pool water through evaporation; and
> 7) initiation of an exothermic oxidation reaction in pools C and D.

LBP-00-19, slip op. at 13. The Board invoked the summary procedures of Subpart K to 10 C.F.R. Part 2, and required the parties to file written presentations and deliver oral argument to determine whether the hearing should go forward. In support of its position, Orange County filed an extensive legal brief and detailed expert report by Dr. Thompson.[11] Dr. Thompson's report presented substantial and material evidence that the probability of an exothermic reaction in the

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10 LBP-00-19, Memorandum and Order (Ruling on Late-Filed Environmental Contentions).

11 *See* Detailed Summary of Facts, Data, and Arguments and Sworn Submission on which Orange County Intends to Rely at Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute with the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant with Respect to the Need to Prepare an Environmental Impact Statement to Address the Increased Risk of a Spent Fuel Pool Accident (November 20, 2000) ("Orange County's Summary re: Contention EC-6"); BCOC Exhibit 1, Declaration of Dr. Gordon Thompson (November 20, 2000) ("Thompson Declaration"); BCOC Exhibit 2, G. Thompson, *The Potential for a Large, Atmospheric Release of Radioactive Material From Spent Fuel Pools at the Harris Nuclear Power Plant: The Case of a Pool Release initiated by a Severe Reactor Accident* (November 20, 2000) ("Thompson Report"). Copies of Orange County's Summary re: EC-6, the Thompson Declaration, and the Thompson Report are provided as attachments to this motion. The County notes that these filings constitute the heart of its Subpart K presentation; and that the only documents not being provided here are the exhibits, which are extremely voluminous. However, the County is prepared to provide the

spent fuel pools, leading to a massive release of radiation from the pools, is foreseeable, and may not be disregarded as a remote and speculative event.

The NRC Staff and CP&L also filed legal and evidentiary presentations, arguing that the probability of a severe spent fuel pool accident is too small to warrant consideration.[12] However, at the oral argument on December 7, 2000, Orange County pointed out that their technical analyses contained considerable omissions and deficiencies, including CP&L's complete failure to address onsite radiation doses that would be incurred as a result of a degraded-core accident with containment failure or bypass; the NRC Staff's oversimplification of onsite radiation doses; and the legal fallacy of assuming, for purposes of avoiding the preparation of an EIS, the acceptability of exposing workers to unlawful radiation levels during the aftermath of a degraded-core melt accident with containment failure or bypass.[13] Moreover, as discussed in Dr. Thompson's December 2000 Declaration at paragraph 18, there are significant areas of technical disagreement between the parties which cannot be resolved without an evidentiary hearing.

A decision from the Licensing Board regarding whether Orange County has satisfied the standard in 10 C.F.R. § 2.1113 for going forward with a full evidentiary hearing is now pending.

Commission with copies of any exhibits that it requests.

12 The significant areas of agreement and disagreement between the parties are described in Dr. Thompson's December 2000 Declaration at pars. 15-18.

13 With respect to this last issue, *see also* Orange County's Summary Re: EC-6 at 31-38.

II. THE COMMISSION SHOULD GRANT REVIEW OF THE NSH DECISION.

The Commission's standard for taking discretionary review is set forth in 10 C.F.R. § 2.786(b)(4). The standard calls for the Commission to give "due weight to the existence of a substantial question" with respect to whether a factual finding is "clearly erroneous or in conflict with a finding as to the same fact in a different proceeding;" whether a "necessary legal conclusion is without governing precedent or is a departure from or contrary to established law;" and whether a "substantial and important question of law, policy, or discretion has been raised." Here, the Staff has made clear and substantial legal and factual errors. Moreover, the Staff's errors in issuing the NSH implicate important issues of law and Commission policy.

A. The Staff Has Failed to Demonstrate That the Criteria of 10 C.F.R. § 50.92(c) Are Satisfied, Or to Address Relevant Evidence That They Are Not Satisfied.

The NRC standard for making a No Significant Hazards determination is found in 10 C.F.R. § 50.92(c)(1)-(3), which provides that the NRC may find that a license amendment poses no significant hazards considerations if it would not:

> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated;
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
> (3) Involve a significant reduction in a margin of safety.

The Staff's No Significant Hazards review is akin to a rulemaking, in the sense that the *Sholly* Amendment requires the Commission to publish notice of proposed NSH determinations and offer an opportunity for public comment.[14] 42 U.S.C. § 2239(a)(2)(B) and (C). In accordance

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14 NSH determinations are also subject to judicial review under the Administrative

with this requirement, the Staff published a proposed NSH determination for the proposed Harris license amendment, and committed to considering "all public and State comments" that are timely received in making its "final determination."[15]

As the U.S. Court of Appeals has recognized, the opportunity to comment on a proposed agency decision is "meaningless unless the agency responds to significant points raised by the public." *St. James Hospital v. Heckler,* 760 F.2d 1460, 1470 (7th Cir.), *cert. denied,* 474 U.S. 902 (1985), *quoting Home Box Office v. FCC,* 566 F.2d 9, 35-36 (D.C. Cir.), *cert. denied,* 434 U.S. 829 (1977) (rule rejected where agency's statement of purpose "provided no indication" of why public criticisms of supporting study were invalid, and "failed to give a reasoned response" to other public comments). The agency must consider "relevant factors," examine the available evidence and articulate a "rational connection" between the evidence and its exercise of discretion. See *Shoreham Co-op Apple Producers v. Donovan,* 764 F.2d 135, 140 (2nd Cir. 1985) and cases cited therein. As demonstrated below, the Staff's NSH Determination is utterly inadequate to meet this test.

1. The Staff Fails to Demonstrate That Issuance of the Proposed License Amendment Would Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

In its discussion of the rationale for the NSH Determination, the Staff reiterates the conclusion of the 1998 Proposed NSH Determination that operation of the Harris facility in accordance with the proposed license amendment would not involve a significant increase in the

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Procedures Act. 42 U.S.C. § 2239(b).

15 *See* Carolina Power & Light; Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Determination, and Opportunity for a Hearing, 64 Fed. Reg. 2,237, 2,239-10 (January 13, 1999).

probability or consequences of an accident previously evaluated. SER at 36. This conclusion is completely unsupported and unjustified, and grossly ignores relevant information in the record.

The Staff's determination has two parts. First, it concludes that the proposed license amendment would not significantly raise the probability of an accident. In apparent response to Orange County's comments that the proposed amendment would double the probability of a fuel handling accident because it would more than double the number of fuel assemblies requiring handling over the remaining life of the Harris plant, the Staff states that the probability of a fuel handling accident is "minimized" to begin with, and that therefore increasing the number of times the fuel is handled would not significantly increase the probability of an accident. This apparently boils down to an argument that multiplying a small number by another number yields a small number. However, the standard relates to the degree of *increase* of the probability, not the overall size of the probability. More than doubling the probability of an accident is a "significant" increment. 10 C.F.R. § 50.92(c)(1). The Staff also completely failed to respond to Orange County's comment that by adding to the electrical load and CCW heat load of existing Harris systems, activation of pools C and D will increase the burden of work on the Harris operators and thereby significantly increase the probability of accidents associated with the reactor's CCW and electrical systems and interruptions in cooling and electricity supply. Orange County's NSH Comments at 5. Thus, the Staff has not demonstrated that the probability of a fuel handling accident would not increase significantly.

Second, the Staff concludes that the consequences of a loss of spent fuel pool cooling event would not be significantly increased from previously evaluated loss of cooling events, because "sufficient time is available for plant operators to take mitigating actions to restore

cooling prior to the pool boiling." SER at 37, 40. To the extent that this rationale is limited to design-basis accidents, the County has no quarrel with it. However, the decision completely ignores the comment submitted by Orange County, that the consequences of a *severe beyond design-basis spent fuel pool accident* would be doubled as a result of doubling the spent fuel pool inventory at Harris. *See* Orange County's NSH Comments at 4. In fact, despite the fact that the potential for a severe spent fuel pool accident is a major consideration of the County's NSH comments and subsequent filings in this proceeding, the Staff's final NSH Determination says *zero* about the issue. This is the height of irrational and arbitrary decisionmaking.

2. The Staff Fails to Show That the Operation of the Facility In Accordance With the Proposed License Amendment Would Not Create the Possibility of a New or Different Kind of Accident From Any Accident Previously Evaluated.

Under its regulations, the NRC may not make a No Significant Hazard determination if it finds that a proposed license amendment would create even the *possibility* of a new or different kind of accident not previously evaluated. The Staff may not rationalize the amendment by addressing the merits of how likely the accident is to occur. *San Luis Obispo Mothers for Peace v. U.S. NRC*, 799 F.2d 1268 (9th Cir. 1986).

In making its proposed NSH Determination in 1998, the Staff asserted that the only kind of accident that might conceivably be considered new is the accidental drop of a fuel rack; but that accident was already considered in relation to pool B. 64 Fed. Reg. at 2,239. Thus, the Staff concluded that the proposed license amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. *Id.* As the County pointed out in its comments, however, there has been no site-specific evaluation of the probability or consequences of severe accidents at pools A and B at Harris, such that the NRC could claim that

the possibility of a spent fuel accident had ever been evaluated at all. Orange County's NSH Comments at 6. In discussing § 50.92(c)(2), the Staff makes no mention of this comment, let alone a response. *See* SER at 38.

Moreover, the Staff fails to acknowledge that the question of whether there is the possibility of a new kind of accident that has not been previously considered has been expressly admitted for litigation by the Licensing Board in this proceeding. *See* LBP-00-19. Now that the issue has been admitted for litigation, and Orange County has met its burden of going forward with substantial material evidence supporting the credibility of a severe spent fuel pool accident[16], it is the Staff's burden to overcome the evidence submitted by Orange County and show that in fact, the accident is not credible. *See Louisiana Energy Services* (Claiborne Enrichment Center), LBP-96-25, 44 NRC 331, 338-39 (1996), *reversed on other grounds,* CLI-98-3, 47 NRC 77 (1998). The Staff has not met this burden in the evidentiary proceeding, and it has not even attempted to meet it in the NSH Determination. Instead, the Staff completely ignores the issue.

Not only has the Staff failed to address or even mention the significance of LBP-00-19, but it has completely ignored the extensive evidence presented by Orange County regarding a credible accident scenario leading to an exothermic reaction in the Harris spent fuel pools and massive radioactive release.[17] The only mention of the ongoing proceeding is the brief recitation

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16 *See* the attached Orange County Summary re: EC-6, Thompson Declaration, and Thompson Report.

17 *See* note 16, *supra.*

of a short procedural history at page 2 of the NSH Determination.[18]

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18 The NSH Determination states as follows:

> On January 31, 2000, BCOC filed four late-filed environmental contentions that challenged the adequacy of the staff's December 21, 1999, environmental assessment related to CP&L's amendment request. On March 3, 2000, the NRC and CP&L responded to the late-filed contentions, and on March 13, 2000, BCOC submitted is reply to the responses. On August 7, 2000, the ASLB issued its Ruling on Late-filed Environmental Contentions. In its ruling, the ASLB admitted one environmental contention (EC-6) regarding the probability of occurrence of BCOC's postulated accident scenario. On November 20, 2000, all parties filed written summaries and on December 7, 2000, the ASLB heard oral arguments related to EC-6.

By reciting the history of the proceeding, the Staff implicitly concedes its relevance. Moreover, the fact that Staff has waited almost two years to issue the NSH Determination demonstrates that a belief by the Staff that the proceeding is relevant to its determination. Yet, the Staff's NSH Determination inexplicably ignores the substantive content of the proceeding. The Staff's brief mention of the procedural history and pleadings seems cynically calculated to create the false and misleading impression that the substance of the proceeding entered into its No Significant Hazards analysis.

3. The Staff Fails to Show That the Operation of the Facility In Accordance With the Proposed License Amendment Would Not Involve a Significant Reduction in a Margin of Safety.

The Staff's conclusion that the operation under the license amendment would not involve a reduction in a safety margin is similarly erroneous. Again, the Staff reaches this conclusion without giving the slightest consideration to the relevance of the pending environmental contention. The central question in the pending environmental proceeding is whether the probability of a severe spent fuel pool accident is significant enough to warrant the preparation of an EIS before issuance of the requested license amendment. Clearly, the question of the probability of a severe accident bears directly on the question of what is the margin of safety of operation. Yet, the Staff makes no attempt to address this issue. Once again, the Staff's analysis is fatally deficient for its failure to address factors that are fundamentally relevant to its determination.

B. The NSH Determination Violates the Requirements of NEPA.

NEPA is the "basic charter for the protection of the environment." 40 C.F.R. § 1500.1(1). Its fundamental purpose is to "help public officials make decisions that are

based on understanding of environmental consequences, and take decisions that protect, restore, and enhance the environment." *Id.* NEPA requires federal agencies to examine the environmental consequences of their actions *before* taking those actions, in order to ensure "that important effects will not be overlooked or underestimated only to be discovered after resources have been committed or the die otherwise cast." *Robertson v. Methow Valley Citizen Council,* 490 U.S. 332, 349 (1989). The primary method by which NEPA ensures that its mandate is met is the "action-forcing" requirement that a "detailed statement," known as an Environmental Impact Statement ("EIS"), be prepared before a federal agency takes any major action which may significantly affect the quality of the human environment. 42 U.S.C. § 4332(2)(C); 40 C.F.R. § 1502.1.

In this case, Orange County has succeeded in putting into contention the question of whether the probability of a severe accident in the Harris spent fuel pools is significant enough to warrant the preparation of an environmental impact statement before the issuance of the license amendment. Orange County has met the Commission's test for gaining the admission of the issue, which is to pose a plausible accident scenario that could lead to significant offsite consequences. The County has also put forth substantial evidence in the Subpart K proceeding which supports the credibility of this scenario. Orange County's evidentiary showing has shifted the burden to the NRC Staff to demonstrate that the scenario is not credible. *See Louisiana Energy Services,* LBP-96-25, 44 NRC at 338-39. To permit the Staff to issue the requested license now, without first defending its decision not to prepare an EIS, would clearly violate NEPA.

C. This Case Raises Important and Novel Legal and Policy Questions.

As discussed above, the key issue in determining whether to require the preparation of an EIS, *i.e.,* the probability of a severe spent fuel pool accident,

encompasses two of the same issues that must be resolved in the No Significant Hazards Proceeding, *i.e.*, whether the proposed license amendment raises the possibility of a new accident that has not previously been considered, or affects the plant's margin of safety.[19] By issuing the license amendment before the completion of the adjudication, the Staff has attempted to hijack the adjudication and substitute its own unilateral and unjustified determination for the participatory decisionmaking process that is now underway before the Licensing Board. The Staff's decision makes a mockery of the license amendment proceeding.

The significance of accident considerations is not a key issue in every licensing proceeding, but it is here. Where the issues raised by a No Significant Hazards review and the licensing review are so intertwined, as they are here, it would be improper and unfair for the Commission to permit the Staff to pre-empt the decision of the Licensing Board by issuing a license before the Board has ruled.

The Commission should also take review in order to protect the integrity of its hearing process and public confidence in this proceeding, which has been seriously undermined by the Staff's action. The Staff's behavior shows a complete disregard for the NRC's hearing process, and a lack of respect for the participants. For reasons that are completely unexplained, the Staff waited almost two full years to issue a Final NSH Determination that differs little from the proposed decision that was published in the Federal Register in 1998. Having waited almost two years, it is all the more inexplicable why the Staff could not have waited for a few more weeks to learn the outcome of the

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19 The third issue, whether the consequences of an accident would be increased, is not a subject of the pending proceeding. However, there can be no legitimate dispute on this issue. The doubling of fuel inventory at Harris would also double the offsite consequences of a severe accident. *See* Thompson December 2000 Declaration, par. 7.

vigorously contested and relevant issues in the pending Subpart K proceeding. After a two year hiatus, it is also curious that Staff suddenly had an urgent need to issue the NSH on the first day of Hanukah and three days before Christmas, when the Staff was no doubt aware that Orange County and its experts were likely to have conflicting holiday plans. The Staff's action gives the NRC the appearance of an agency that lacks respect for its own adjudicatory process or the members of the public who in good faith submit their concerns to resolution through that process, and who devote substantial time and resources to adjudicating a meaningful result.[20]

III. ORANGE COUNTY SATISFIES THE COMMISSION'S REQUIREMENTS FOR ISSUING A STAY.

Orange County requests the Commission to stay or suspend the issuance of the Harris license amendment pending the outcome of the pending adjudication, including any hearing that may be ordered by the Licensing Board. In addition, Orange County requests the Commission to stay or suspend the issuance of the license amendment until completion of any EIS that should be required by the Board. This requests satisfies the Commission's standard for a stay, as set forth in 10 C.F.R. § 2.788(e).

First, Orange County has made a strong showing that it is likely to prevail on the merits. Because the Staff has not demonstrated satisfaction of the criteria in 10 C.F.R. § 50.92, the NSH Determination must be reversed. *See San Luis Obispo Mothers for Peace*

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20 Notably, the Staff holds itself out as "the Commission" throughout the SER and the NSH Determination. This leaves members of the public, who are generally unschooled in the minutae of NRC procedures, with the unmistakeable impression that the issuance of the license amendment is the decision of the Commissioners themselves, who also happen to sit as the ultimate judges over the outcome of the adjudicatory proceeding. Thus, to the average member of the public it must certainly appear that the Commission has pre-empted its own adjudication of issues that are central to both the No Significant Hazards determination and the question of whether an EIS must be prepared prior to license issuance.

v. NRC, 799 F.2d at 1271 (NSH determination reversed where criteria of 10 C.F.R. § 50.92 were not satisfied). Moreover, the Staff has completely failed to meet the Administrative Procedure Act's requirement that agency decisions must be explained and address all relevant factors, including comments by the public. In fact, the NSH Determination is a particularly egregious example of an agency's disregard for significant factors which are directly within its purview and which bear directly on its decision. Finally, as discussed above, the decision violates NEPA because it circumvents the duly-instituted legal process for determining whether a severe spent fuel pool accident is likely enough to warrant the preparation of an EIS. If the license amendment is allowed to stand, the public will be subjected to an increased risk of a severe spent pool accident, without any justification by the NRC Staff as to why the potential for such an accident may be ignored.

Second, the citizens of Orange County and other members of the public will be irreparably injured by the increased risk of a severe spent fuel pool. As set forth in the attached Thompson December Declaration, activation of pools C and D would create the potential for a large release of radioactive material from pools C and D into the Harris fuel handling building and from there to the surrounding atmosphere. *Id.*, par. 3. Radiation from the deposited material would cause irreparable harm to the affected environment, to persons who inhabit the affected environment, and to persons who consume food or water from the affected environment. *Id.*, par. 4. This potential for a large release would arise from CP&L's proposal to use high density racks in pools C and D. *Id.*, par. 5. A variety of events could lead to a loss of water from pools C and D, including a degraded-core accident at the Harris reactor with containment failure or bypass. *Id.*, par. 6. This type of accident is a foreseeable event, based on information

available from the NRC Staff and CP&L. *Id.*, pars. 8-9. The consequences of such an accident would be as much as 2.5 times greater than the consequences of a spent fuel pool accident involving the current inventory of spent fuel at Harris, given that CP&L would be more than doubling its inventory of spent fuel. *Id.*, par. 7.

Moreover, if CP&L is permitted to incur transportation costs involved in bringing spent fuel from other reactors to store in high density racks in pools C and D, these expenditures may later be treated as "sunk costs" if an EIS is prepared, thus prejudicing the decisionmaking process against other alternatives. *See, e.g., Public Service Company of New Hampshire* (Seabrook Station, Units 1 & 2), ALAB-422, 5 NRC 503, 532 (1977) (acknowledging the difficulty of determining whether time and money already expended at the time of an environmental analysis should weigh in favor of the proposal).

Fourth, neither the NRC Staff nor CP&L will suffer irreparable injury as a result of a stay or suspension of the license amendment. As a regulator, the Staff is not in a position to suffer harm. CP&L will not suffer irreparable harm, even though it is running out of core off-load space in pools A and B. Under appropriate restrictions, it would be acceptable to use pools C and D for low-density storage of spent fuel, pending the outcome of the adjudicatory proceeding on Contention EC-6. *See* Thompson December Declaration, pars. 12, 13, and 14.

Fifth and finally, the issuance of a stay is in the public interest. As discussed above, this case involves a NEPA proceeding in which the central contested NEPA issue is essentially the same as the disputed No Significant Hazards issue. Orange County, which has responsibility for the health and welfare of thousands of citizens, has intervened in the adjudication for the purpose of resolving its concerns about the safety and environmental impacts of the massive increase in the volume of spent fuel which

CP&L proposes to store in high-density racks. It is fair and appropriate to protect the integrity of this ongoing proceeding and public confidence in the NRC's participatory decisionmaking process, by delaying the issuance of a license until the contested NEPA issues have been resolved.

IV. CONCLUSION

For the foregoing reasons, the Commission should take review of the NSH Determination and immediately suspend its effectiveness until completion of the adjudicatory proceeding or the completion of an Environmental Impact Statement, whichever event is later.

Respectfully submitted,

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December 22, 2000

Excerpts from Transcript of Oral Argument in Subpart K Proceeding (December 7, 2000):

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