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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF RESPONSE TO INTERVENOR'S REQUEST FOR ADMISSION OF LATE-FILED ENVIRONMENTAL CONTENTIONS

I. INTRODUCTION

Pursuant to an "Order (Granting Amended Request for Time Extension to File Reply)," issued on February 14, 2000, by the Atomic Safety and Licensing Board ("Board"), and 10 C.F.R. § 2.714(c), the staff of the Nuclear Regulatory Commission ("Staff") hereby responds to "Orange County's Request for Admission of Late-Filed Environmental Contentions," dated January 31, 2000 ("Request for Admission"). As discussed below, none of Orange County's ("Orange County" or "BCOC") proposed contentions is admissible. Therefore, Orange County's Request for Admission of its late-filed contentions should be denied.

II. DISCUSSION

A. Late-Filed Contentions

The admissibility of any late-filed contention, including those filed on subsequent NRC environmental review documents, is governed by the criteria set forth in

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10 C.F.R. § 2.714(a)(1)(i)-(v). *Sacramento Municipal Utility Dist.* (Rancho Seco Nuclear Generating Station), CLI-93-12, 37 NRC 355, 363 (1993). The proponent of the admission of late-filed contentions bears the burden of demonstrating that a balancing of these factors weighs in favor of admission of the proposed contentions. *Cf. Texas Util. Elec. Co.* (Comanche Peak Steam Electric Station, Units 1 & 2), CLI-92-12, 36 NRC 62, 69 (1992) (petitioners for late intervention bore burden to demonstrate that balancing of factors weighed in favor of their intervention).

The institutional unavailability of a licensing-related document does not establish good cause for filing a contention late if information was publicly available early enough to provide the basis for the timely filing of that contention. *Duke Power Co.* (Catawba Nuclear Station, Units 1 & 2), CLI-83-19, 17 NRC 1041, 1045 (1983). The Commission has not held that a showing that the Staff's environmental review documents significantly differ from the applicant's environmental report is always necessary to raise a good contention. *Rancho Seco*, CLI-93-12, 37 NRC at 363. Without such a showing, an intervenor may be able to meet the late-filed contention requirements of 10 C.F.R. § 2.714(a)(1) by presenting significant new evidence not previously available. *Id.* In addition, an intervenor's lateness may not be fatal if its argument rests significantly on a licensee document prepared after the submission of the original contention and the intervenor brought the argument promptly to the Board's attention. *See Yankee Atomic Elec. Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 255 (1996).

Because the proffered contentions are being submitted beyond the time limit specified in 10 C.F.R. § 2.714, BCOC must address the five factors for late-filed contentions discussed above and establish that consideration of the factors weighs in favor of admission.

BCOC's contentions were to be filed on April 5, 1999, and BCOC timely filed five environmental contentions, but the Board dismissed them without prejudice to file at an appropriate time. BCOC asserts that the issuance of the Staff's EA provided an appropriate time to file these contentions. BCOC filed its environmental contentions 46 days after receipt of the EA. Good cause is predicated on the specific circumstances of this case -- BCOC previously filed timely environmental contentions which were dismissed without prejudice; and BCOC received the EA on December 16, 1999, while counsel was preparing a voluminous pleading to be filed in this proceeding. Based on the above circumstances of this case, the Staff does not object on the basis of timeliness, except to the extent that BCOC is raising a security issue related to its contention regarding sabotage. Such an issue is not an environmental issue based upon the EA and could have been raised within the time limits. That issue is, therefore, late without good cause.

As to the remaining factors, the Staff agrees that there is no other forum for seeking the relief requested and that there are no other parties to represent the interests of BCOC. But the Staff does not agree that BCOC's participation may be expected to assist in the development of a sound record. BCOC's contentions are supported by the report of Dr. Gordon Thompson (Thompson Report), who is offered as an expert. Neither the Thompson Report nor the Request for Admission provides a basis for admission of any of the four contentions. Moreover, the submissions consist of discussions of severe accidents and their consequences that are not required to be analyzed in conjunction with a change to the spent fuel pools, and, therefore, raise concerns which are irrelevant to this proceeding. Therefore, BCOC's participation will not assist in the development of a sound record. Finally, there is no doubt that BCOC's participation will broaden the issues and delay the proceeding.

Since BCOC has demonstrated good cause for failure to file on time, the most important of the five factors,[1] except as noted above, and has met two of the other four criteria of 10 C.F.R. § 2.714(a)(1), BCOC's late-filed contentions should not be dismissed because of their lateness. As noted below, however, for other reasons, none of the contentions should be admitted.

B. Legal Standards for Admission of Contentions

1. Standards Applicable to All Contentions

In order for a contention to be admitted to a proceeding, the requirements of 10 C.F.R. § 2.714 must be met. *Duke Energy Corp.* (Oconee Nuclear Station, Units 1, 2, & 3), CLI-99-11, 49 NRC 328, 333 (1999); *Yankee Atomic Elec.Co.* (Yankee Nuclear Power Station), CLI-96-7, 43 NRC 235, 248 (1996). A contention must meet the standards set forth in 10 C.F.R. § 2.714(b)(2), which provides that each contention must consist of a "specific statement of the issue of law or fact to be raised or controverted" and must be accompanied by:

(i) A brief explanation of the bases of the contention;

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[1] *See Commonwealth Edison Co.*(Braidwood Nuclear Power Station, Units 1 and 2), CLI-86-8, 23 NRC 241, 244 (1986).

> (ii) A concise statement of the alleged facts or expert opinion which supports the contention . . . together with 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;
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> (iii) Sufficient information . . . to show that a genuine dispute exists with the applicant on a material issue of law or fact.

10 C.F.R. § 2.714(b)(2). The failure of a contention to comply with any one of these requirements is grounds for dismissing the contention. 10 C.F.R. § 2.714(d)(2)(i); *Arizona Public Serv. Co.* (Palo Verde Nuclear Generating Station, Units 1, 2 & 3), CLI-91-12, 34 NRC 149, 155-56 (1991). When a postulated accident scenario provides the premise for a contention, a causative mechanism for the accident must be described and some credible basis for it must be provided. *See Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29, 44 (1989), *remanded on other grounds*, CLI-90-4, 31 NRC 333 (1990).

In order for a dispute to involve a material issue of law or fact, its resolution must make a difference in the outcome of the proceeding. *Oconee*, CLI-99-11, 49 NRC at 333-34, *citing Final Rule, Rules of Practice for Domestic Licensing Proceedings -- Procedural Changes in the Hearing Process*, 54 Fed. Reg. 33,168, 33,172 (1989). *See also* 10 C.F.R. § 2.714(d)(2)(ii) (a contention must also be dismissed where the "contention, if proven, would be of no consequence . . . because it would not entitle [the] petitioner to relief."). Moreover, contentions that are not supported by some alleged fact or facts should not be admitted nor should the full adjudicatory hearing process be triggered by contentions that lack a factual and legal foundation. *Oconee*, CLI-99-11, 49 NRC at 334-35, *citing* 54 Fed. Reg. at 33,170.

2. Standards for Admission of Environmental Contentions

All of BCOC's late-filed proposed contentions relate to environmental issues, and many of the bases for these proposed contentions involve severe accidents. The Commission has laid down standards, in addition to those described above, applicable to the admission of such contentions, and for the treatment of environmental contentions involving severe accidents. These standards are set forth below.

The National Environmental Policy Act of 1969, as amended ("NEPA"), is to be interpreted by a "rule of reason." *See Vermont Yankee*, ALAB-919, 30 NRC at 44, citing *Limerick Ecology Action, Inc. v. NRC*, 869 F.2d 719, 739 (3d Cir. 1989) and *San Luis Obispo Mothers for Peace v. NRC*, 751 F.2d 1287, 1300 (D.C. Cir. 1984), *aff'd en banc*, 789 F.2d 26, *cert. denied* 479 U.S. 923 (1986). If a contention claims that an EIS is necessary or inadequate in some respect, the "rule of reason" provides that agencies need not consider "remote and speculative risks" or "events whose probabilities they believe to be inconsequentially small." *Id.* In addition, neither NEPA nor the case law based thereon requires a "worst case analysis." *See Vermont Yankee*, ALAB-919, 30 NRC at 44, citing *Robertson v. Methow Valley Citizens Council*, 490 U.S. 332, 333-34 (1989).

The Commission's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 Fed. Reg. 32,138 (1985)("Severe Accident Policy"), addresses, among other things, consideration of severe accidents in environmental impact statements (EISs) prepared in the Staff's review of initial operating licenses. *Vermont Yankee*, ALAB-919, 30 NRC at 50 n.29. It does not require the Staff to consider such matters where no EIS is required. In addition, the Commission considers the environmental risks of beyond design-basis accidents in initial operating license proceedings as a matter of discretion, rather than as a requirement of NEPA. *Vermont Yankee*, ALAB-919, 30 NRC at 50, n.29, *citing San Luis Obispo Mothers for Peace*, 751 F.2d at 1301. The Atomic Safety and Licensing Appeal Board ("Appeal Board") found no Commission intent to extend that discretionary policy to a license amendment proceeding. *Vermont Yankee*, ALAB-919, 30 NRC at 50-51 n.29. That policy should not be extended to this proceeding.[2]

C. BCOC's Contentions Do Not Meet the Standards for Admission of Contentions Set Forth in 10 C.F.R. § 2.714

> CONTENTION EC-1: 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 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.

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[2] In *Limerick*, the Court of Appeals for the Third Circuit held that the Severe Accident Policy Statement was entitled to no deference, and the Commission could not rely on it to exclude the consideration of severe accident mitigation design alternatives ("SAMDAs") in an EIS prepared pursuant to its "Statement of Interim Policy, Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969," 45 Fed. Reg. 40,101 (1980)("Interim Policy Statement"). *Limerick*, 869 F.2d at 731-36. The court in *Limerick*, however, did not examine whether severe accidents should be considered with respect to amendments. Indeed, the Commission, in the Interim Policy Statement, directed only that severe accidents be considered in EISs prepared with respect to construction permits and initial operating licenses. Interim Policy Statement, 45 Fed. Reg. at 40,103.

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[3], 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.

In support of this contention, BCOC proffers six bases. The bases are designated A through F. Basis F has two subparts. Each of the bases and subparts, and the reasons why they are inadequate to support admission of proposed Contention EC-1, are set forth below.

Basis A. The NRC is required, pursuant to NEPA, 42 U.S.C. § 4332(C), and 10 C.F.R. § 51.20(a), to prepare an EIS for CP&L's proposed action because it is a "'major federal action significantly affecting the quality of the human environment.'" Request for Admission at 3-4. Further, if portions of a proposed action have been previously addressed in an EIS, a new EIS is required if a major federal action will occur and there is new

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

information showing that there will be a significant effect on the human environment not previously considered. *Id.* at 4.

Staff Response to Basis A. This argument does not provide an adequate basis for admission of this contention. It merely recites the regulatory requirements and makes the unsupported conclusion that the proposed action is a "major federal action significantly affecting the quality of the human environment," citing no facts or expert opinion. Moreover, it does not recite a material fact in issue.

BCOC provides absolutely no support for its allegation that the proposed amendment constitutes a major federal action significantly affecting the quality of the environment. In fact, neither NEPA nor the Commission's regulations require the preparation of an EIS in this case. The Commission's regulations at 10 C.F.R. § 51.20(b) list actions requiring an EIS; 10 C.F.R. § 51.22 lists actions eligible for categorical exclusion or otherwise not requiring environmental review; and 10 C.F.R. § 51.21 states that actions not falling within either § 51.20(b) or § 51.22 require an EA. The issuance of an amendment authorizing spent fuel pool storage capacity expansion is not listed as an action requiring an EIS and BCOC has raised nothing in this basis that would indicate that an EIS is required in this case. BCOC fails to satisfy 10 CFR §2.714 (b) (ii), which requires a contention to be supported by the alleged fact or expert opinion which supports the contention together with references to those specific sources and documents on which petitioner intends to rely to establish those facts or expert opinion.

Basis B. The proposed action will result in "substantial changes to the physical characteristics and mode of operation" of Harris, due to 1) the increase in the number of spent fuel assemblies permitted to be stored and the increase in the amount of radioactive material, and 2) the reliance on administrative measures over physical measures to prevent criticality in pools C & D. *Id.* at 4-5.

Staff Response to Basis B. This basis does not provide support for the contention., or even offer information demonstrating a genuine dispute. BCOC merely alleges that there will be substantial changes because of the increase in the number of fuel assemblies and the reliance on administrative measures to prevent criticality.[4] BCOC provides no support for these conclusions in the form of facts or expert opinion. BCOC therefore fails to satisfy 10 CFR §2.714 (b) (ii), which requires a contention to be supported by the alleged fact or expert opinion together with references to those specific sources and documents on which petitioner intends to rely to establish those facts or expert opinion. Therefore, this basis must be dismissed.

Basis C. There is a need to evaluate the "extent to which the proposed amendment would create an additional or incremental risk of accidents"; and "previous environmental analyses of the existing operation, and determine whether they are adequate to address the incremental risk posed by the proposed license amendment." *Id.* at 5-6.

Staff Response to Basis C. This basis does not support admission of the contention because BCOC offers no factual, expert or documentary support. BCOC presents no basis for concluding that there is any incremental risk posed by the proposed amendment, and thus, fails to satisfy 10 CFR §2.714 (b) (ii). BCOC states that the evaluation of the proposed amendment should be accomplished by comparing the risk of the existing operation with the

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[4] The Staff discusses criticality further in connection with Basis F.2.c, below.

risk of the operation as proposed, but provides no information as to how to conduct the evaluation, other than to say that "risk" is not based "as is sometimes done" on probability times consequences but is rather "the potential for an accident, encompassing both the probability and consequences." Request for Admission at 5-6, n. 3. BCOC does not provide a reference for this definition or otherwise explain its meaning or its departure from the commonly accepted definition.

Basis D. The NRC's evaluation of accident risk, contained in the 1983 EIS for the operating license (NUREG-0972), CP&L's Individual Plant Examination (IPE) of 1993, and CP&L's Individual Plant Examination for External Events (IPEEE) of 1995 did not evaluate spent fuel accidents. The findings contained in NUREG-0575, the Generic Environmental Impact Statement on the handling and storage of spent fuel, regarding the risk of spent fuel pool accidents are no longer applicable, because new information demonstrates that "the risks of a severe spent fuel accident during high-density pool storage of spent fuel are significant." Request for Admission at 6-7.

Staff Response to Basis D. This basis fails to satisfy 10 CFR §2.714 (b) (ii), which requires a contention to be supported by the alleged fact or expert opinion which supports the contention together with references to those specific sources and documents on which petitioner intends to rely to establish those facts or expert opinion. Basis D concludes that new information demonstrates that "the risks of a severe spent fuel accident during high-density pool storage of spent fuel are significant," but does not specify what the new information is or where it may be located. Basis D appears to be simply a restatement of

Bases C and F. In addition, BCOC's complaint that the EIS for Harris issued in 1983[5] did not evaluate spent fuel pool accidents does not support admission of its contention because, as explained above, the Severe Accident Policy applies only to reactors, and does not require consideration of severe accidents with respect to spent fuel pools.[6]

> Basis E. CP&L has not evaluated the increment of accident risk that would arise from operation of SFPs C and D.

Staff Response to Basis E. This basis does not raise a litigable issue, contains no factual assertions, and otherwise does not support admission of this contention. It fails to satisfy 10 CFR §2.714 (b) (ii), because, although BCOC asserts that the staff has not performed an analysis, it provides no support for its contention, via facts or expert opinion based on credible facts, that such an analysis is in fact required or will result in relevant findings. Therefore, Basis E must be dismissed.

Basis F. BCOC alleges that the EA is incorrect in its evaluation of the increment of accident risk that would arise because: (1) new information indicating that the risk for high density pool storage is significant and not properly evaluated in the GEIS (NUREG-0575), the EIS for Harris or the EA (Request for Admission at 7); (2) the increment of accident risk from operation of pools C and D would be significant, "by itself and in comparison to the

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[5] NUREG-0972, "Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant Units 1 and 2, Docket Nos. STN 50-400 and 50-401, Carolina Power and Light Company (October 1983).

[6] The argument in the Thompson Report that a PRA analysis should be done to address the risks of accidents in the Harris spent fuel pools is without merit and lacks a basis in fact or law. *See* Thompson Rep. at 6; Thompson Rep. Appendix B. A PRA analysis is neither justified nor required in this case. *See* "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," 60 Fed. Reg. 42,622, 42,624-25, 42, 628 (1995).

baseline accident risk . . .[and the] increment of risk arises from the proposed changes in the physical characteristics and mode of operation of the Harris plant, specifically the greatly increased inventory of spent fuel permitted at the Harris site and the significant weakening of criticality prevention measures" (*Id.* at 7-8.); and (3) the Staff failed to take new information regarding risk of sabotage into account.

Staff Response to Basis F. In support of this basis BCOC states, in a footnote, that its concern is the increment of risk posed by operation of pools C and D. Request for Admission at 7-8 n. 5. However, in order to evaluate that risk, BCOC believes that the risk of operation prior to the amendment, that is with only pools A and B in operation, must be assessed as a baseline for comparison. *Id.* BCOC states, "if an adequate evaluation of the risk posed by pools A and B became available, this evaluation would shed light on the cumulative impacts of the proposed license amendment, but would not affect the significance of the increment of risk that would arise from that amendment." *Id.* BCOC appears to be saying that even without knowing what the "baseline" is, it knows that the incremental risk is significant. BCOC repeatedly refers to "new information"; however, it fails to identify any information that allows it to conclude that "the increment of risk" is significant without establishing what that increment is or that it, in fact, exists. Since "risk" for Orange County is not the familiar definition of probability times consequences but is rather some unexplained "potential," it is not possible to say with any certainty what the County's concerns are.

Throughout this basis and the other bases, BCOC refers to the Thompson Report and the new information contained therein. A careful reading of the report demonstrates that it does not contain new information regarding the issues raised by BCOC and that there are significant inaccuracies and lack of bases for many of the conclusions reached.[7]

In Basis F-1, BCOC refers to new information that is "summarized" in the Thompson Report. This "new information" allegedly shows that an accident involving exothermic reaction of zircaloy fuel cladding could contaminate land with Cesium-137 to the extent that relocation of populations could be required over an area as large as North Carolina. Request for Admission at 8. This "new information" is not identified either in the Request for Admission or in the Thompson Report. The footnote on page 9 of the Request for Admission reveals that a degraded core accident is evaluated in the EIS for the Harris OL and in Harris's IPE, but that these reports do not discuss the effect of a reactor accident on the operation of the fuel pools at Harris. Request for Admission at 9 n.6.

BCOC faults the Staff's EA for not discussing the alleged new information in Appendix C of the Thompson Report, to wit: that the loss of water in the Harris pools is an almost certain outcome of a degraded core accident with containment failure or bypass.

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[7] Some examples of the lapses in the report follow. As stated elsewhere in this brief, the conclusion that partial or total loss of water in the SFPs would be "an almost certain outcome" of a severe reactor accident involving containment failure is totally unsupported. In discussing the effects of earthquake and cask drop, the report refers to studies of the Robinson, Millstone and Ginna plants, but contains no assessment of the effects at Harris. Appendix C at C-2 to C-4. The report also makes the unsupported assumption that if the postulated reactor accident occurs, SFP cooling would cease and would not resume. *Id.* at C-5. The report relies on an outdated figure for maximum heat load in pools C & D (15.6m BTU/Hr), ignoring the fact that the heat load will be limited by technical specifications to 1m BTU/Hr. *Id.* The estimate of the upper bound of temperature rise is based on a value applicable to fuel aged 1 year, which would be inapplicable to the fuel that will be stored in pools C & D. Appendix D at D-3 to D-4. The conclusion that fuel aged in excess of 10 years is subject to exothermic reaction is without basis. *Id.* at D-5. *See* NUREG-0649 at 75. No effort is made in the report to relate any of the analyses discussed to the Harris SFPs. *Id.* at D-7.

Request for Admission at 9. Yet, Appendix C contains no new information, other than the unsupported conclusions of Dr. Thompson, and contains no analysis or basis for his conclusion that the loss of water is "an almost certain outcome" of the postulated accident. *See* Request for Admission at 10, 11; Appendix C at C-5. *See also* Thompson Rep. at 8, 13-14; Appendix B at B-6. Nor does it contain any specific references to "other literature" constituting "new" information, which supports this conclusion. In addition, Appendix C states that "[a] comprehensive application of PRA techniques to the Harris fuel pools is a task beyond the scope of the author's present work for Orange County." Appendix C at c-2. Thus, it appears that the number the Thompson Report supplies for the probability of an exothermic reaction leading to the need to evacuate by all of North Carolina (that number is 1) is based not on a probabilistic risk assessment or any other assessment but on mere speculation. Dr. Thompson states, "it can be assumed that pool cooling would cease during the accident. And would not resume." Appendix C at C-5. This assumption is not based on any facts or analysis. BCOC introduced no "new information," but rather conclusions unsupported by facts or credible analysis. BCOC has not provided a sufficient basis for the contention that an EIS is required in order to analyze the postulated series of accidents. As the Appeal Board stated in *Vermont Yankee,* "when a postulated accident scenario provides the premise for a contention, a causative mechanism for the accident must be described and some credible basis for it must be provided." *Vermont Yankee*, ALAB-919, 30 NRC at 44. Here, BCOC does not specify the cause of the initiating accident - core degradation with containment bypass or failure. More importantly, BCOC fails to state a basis, either in the Request for Admission or from the Thompson Report, for its conclusion that the loss of water to the spent fuel pools and exothermic reaction in the pools is the certain result of the postulated reactor accident.[8] Nor has BCOC demonstrated that this postulated accident series is a design basis accident. Therefore, the basis is not admissible in support of the contention.

BCOC has not demonstrated that the accident with which it is concerned is an accident which is within the scope of this proceeding, that is, that it is not a remote and speculative accident that need not be considered in connection with the proposed amendment. Such remote and speculative occurrences are excluded from consideration by NEPA's "rule of reason," discussed above. *See also Pacific Gas & Elec. Co.* (Diablo Canyon Nuclear Power Plant, Units 1 & 2), ALAB-880, 26 NRC 449, 458 (1987); *Public Service Electric & Gas Co.* (Salem Nuclear Generating Station, Unit 1), ALAB-650, 14 NRC 43, 48 n. 5, 62-3 n.29 (1981). In fact, BCOC has produced nothing to demonstrate the probability of its spent fuel pool accident, other than Dr. Thompson's unsupported conclusion that it is an almost certain result of the degraded core accident. In Appendix B of the Thompson Report, the probability of degradation of the reactor core is addressed, based upon the analysis contained in CP&L's IPE and IPEEE.[9] Thompson Rep., Appendix B at B-4-7. The

[8] The Thompson Report correctly points out that none of the NRC documents cited in the report or appendices provide support for the conclusion that the postulated spent fuel pool accident will "almost certainly" follow the postulated reactor accident. *See e.g.* Thompson Rep. at 6, Appendix B at B-7, Appendix C.

[9] It should be noted that at no point in Appendix C does the author point to the specific pages or sections of the IPE and the IPEE to which he refers. In fact, neither the Thompson Report, nor BCOC's Request for Admission provide page numbers for any reference. They merely make a general reference to the documents, leaving the other parties and the Board to actually locate the analyses to which they refer. This is impermissible pursuant to Commission precedent. *See Commonwealth Edison Co.* (Zion Nuclear Power Station, Units 1 & 2), CLI-99-4, 49 NRC 185, 194 (1999), *aff'd Dienethal v. NRC,* No. 99-1132 (D.C.Cir.

(continued...)

report states that this figure is 7x10-5. *Id.* at B-4. The probability of containment failure leading to a release in the RC-5 category is placed at 3x10-6. The report concludes that, based upon an alleged IPE prediction that 15% of core damage sequences will lead to *significant* degree of containment failure, there is a total probability of 1X10-5. *Id.* at B-6-7. There is no indication of how this figure was reached. In fact, the figure is misleading. The IPE actually indicates that the conditional probability of *all* releases, significant or not, is 15%. *See* NRC Staff's Evaluation of the Shearon Harris Nuclear Plant Individual Plant Examination (IPE Submittal), "Technical Evaluation Report of the Shearon Harris Individual Plant Examination Back-End Submittal," (ERI/NRC 95-103) at viii, 21 (1995). (Exhibit A). Table E.1 in ERI/NRC 95-103 shows that a certain percentage of the releases would be insignificant. For example, 3.2% of containment failure modes consist if containment failures with in-vessel recovery, prior to vessel breach. *Id.* at vi. Therefore, the conclusion that 15% of the releases would be significant is not supported by the documentation referenced.

In addition, the Thompson Report does not attempt to evaluate the probability that any of the containment failure scenarios will affect or preclude entry into the fuel handling building. The report merely concludes, without support, that if there is an RC-5 category release "the Harris plant and its immediate surroundings would become radioactively contaminated to the point where access by personnel will be precluded. Accidents in other release categories would release smaller amounts of radioactive material, but could also

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[9](...continued)
Jan. 21, 2000) (*per curiam*)..

contaminate the Harris plant to the point where access by personnel would be precluded." Thompson Rep. at B-6. [Moreover, the Report does not consider meteorological effects (e.g. wind direction), type of containment failure, or other factors which would affect the probability that the postulated core degradation with containment bypass or failure would pose a credible and sustained threat to access to the fuel handling building.] *See, e.g.*, ERI/NRC 95-103 at 21. If those factors had been considered, the probability would be far less than postulated in the Thompson Report.

In *Vermont Yankee*, the Appeal Board rejected, as remote and speculative, contentions premised on a severe accident involving a self-sustaining cladding fire in a spent fuel pool. *Id.* at 45-47, 50-52, *remanded for further findings*, CLI-90-4, 31 NRC 333, *clarification requested*, ALAB-938, 32 NRC 154, *clarified and dismissed*, CLI-90-7, 32 NRC 129 (1990). In that case, involving a spent fuel pool expansion, the Appeal Board denied admission of a contention similar to the one offered herein. The intervenor in that proceeding submitted a contention focusing on:

> an unspecified, hypothetical reactor accident involving hydrogen generation, failure of the Mark I containment, and hydrogen detonation in the reactor building, which also houses the spent fuel pool. This accident in turn allegedly would threaten the pool cooling water systems or pool structure itself, leading to pool heatup and ultimately a zircaloy cladding fire.[10]

[10] The contention asserted that the EA failed to consider the consequences and risks posed by the hypothetical accident, which would result in risks greater than those previously evaluated in connection with the reactor. The risk was sufficient to render the proposed amendment a "major federal action significantly affecting the environment," therefore requiring an EIS. *Vermont Yankee,* ALAB-919, 30 NRC at 52.

Vermont Yankee, ALAB-919, 30 NRC at 42. In addressing the admissibility of the contention, the Appeal Board stated:

> It should go without saying that reactors and spent fuel pools are not expected to have accidents, or a series of accidents, like that set forth in this contention. . . . Further, spent fuel pools must be designed "to prevent significant reduction in fuel storage coolant inventory under accident conditions." 10 C.F.R. Part 50, Appendix A, General Design Criteria 61. Therefore, 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 -- (1) a severe hydrogen-generating and detonating reactor accident that somehow leads to (2) a gross loss of spent fuel pool water and subsequent zircaloy fire. In other words, the two accidents at the heart of the contention are individually among the worst things that can even be hypothesized for a reactor and an spent fuel pool, respectively, in terms of potentially significant offsite consequences for the public.

Id. at 43 (footnote omitted).

BCOC cites the Reactor Safety Study ("WASH-1400") and NUREG-1353[11] in support of its thesis that its proffered severe accident scenario is not remote and speculative. It quotes the Executive Summary of NUREG-1353 in support of this proposition, but the quote does not support its thesis. Request for Admission at 11. WASH-1400 shows the risks of beyond design basis accidents in spent fuel pools as "orders of magnitude" below those involving the reactor core. NUREG-1353 concluded that the probability of a zircaloy cladding fire[12] resulting from the loss of water was estimated to have a mean frequency value of 2x10-6. The risks and consequences of a spent fuel accident were found to meet the

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[11] E.D. Throm, NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (April 1989).

[12] The postulated exothermic air reaction is the same as the zircaloy cladding fire discussed in *Vermont Yankee.*

objectives of the Safety Goal Policy Statement. It also concluded that spent fuel three years out of the reactor could be air cooled, even in high density racks. *Id.* At 1-1. The report further found that zircaloy fires would not propagate to PWR fuel stored in high density racks if the fuel had an approximate decay time of 730 days (2 years). *Id.* at 4-12. Therefore, it is not credible that an exothermic reaction would propagate or even occur in pools C and D due to the age of the fuel that CP&L proposes to place in those pools - 5 years out of the reactor.[13] Pools C and D are approximately 300 feet from pools A and B and BCOC and BCOC has not provided a basis for the proposition that an exothermic air reaction could propagate from A or B to C or D.

Basis F-2 addresses the "significant increment of accident risk posed by license amendment," and gives the following reasons to support its contention that the proposed amendment significantly increases the risk of an accident: 1) increased fuel storage capacity; 2) higher density storage; and 3) weakening of criticality prevention measures. None of these reasons supports the allegation that a significant increase in risk is posed by the proposed amendment.

As regards increased fuel storage capacity, BCOC believes that because pools C and D would add the capacity for storage of 4,715 additional G21

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[13] As noted in the Thompson Report, CP&L intends to install racks in pool C in three campaigns (2000, 2005 and 2014), and in pool D in two campaigns (2016 and a date to be determined). Thompson Rep. at 3. Therefore, large portions of the pools will remain empty for many years, thus excluding the possibility that the pools would reach the boiling point even with extended loss of cooling, due to the additional heat sink. In addition, as the years progress, the spent fuel in the pools would age, so that when the pools are filled to capacity, the bulk of the fuel would be 9 to 25 years out of the reactor, lowering the probability of boiling and propagation of an exothermic reaction.

assemblies at Harris, an accident at these pools could release to the atmosphere a substantial fraction of the inventory of Cesium-137 in these pools. The Request for Admission points to the Thompson Report at Appendices D and E as support for this proposition. However, although Dr. Thompson states that for "scenarios which involve partial uncovery of fuel, the reaction could affect fuel aged 10 or more years," he offers no authority to support this conclusion. Dr. Thompson's is the only opinion of which the Staff is aware that holds that fuel five years or more out of the reactor is susceptible to zircaloy fire/exothermic reaction. *See, e.g.*, NUREG/CR-0649, Spent Fuel Heatup Following Loss of Water During Storage, at 85-87 (1979) (Exhibit B).

Moreover, NUREG-0972, the Final Environmental Statement related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2 (1983)("FES") and NUREG-1038, the Safety Evaluation Report related to the Operation of Shearon Harris Nuclear Plant, Units 1 and 2 (1983) ("SER"), both evaluated operation of Harris as a two unit facility with four fuel pools. *See, e.g.*, SER at §§ 9.1.1, 9.1.2 (Exhibit C); FES at § 5.9.4 (Exhibit D). Therefore, there is no significant incremental increase in risk of accident, due to the proposed changes in the physical characteristics and mode of operation, not already evaluated for pools C and D. The second reason BCOC gives as support for its contention of significant increase in risk is higher density storage. BCOC says, "[o]ther factors being equal, this reduced distance [between assemblies] would increase the propensity of pools C and D . . . to experience an exothermic reaction of fuel cladding in the event of partial or total loss of water." Request for Admission at 12. However, other factors are not equal. The fuel to be stored in the C and D pools will be at least five years out of the reactor. Dr.

Thompson's belief that such fuel is susceptible to exothermic reaction does not appear to be based on the scientific literature. *See, e.g.,*Exhibit C at 73-77, 85-87.

With respect to criticality, BCOC simply asserts that the proposed amendment results in "significantly increasing the probability that a criticality accident would occur at the Harris plant." Request for Admission at 14. As explained above, however, BCOC does not provide any baseline probability of such an accident's occurring, and, indeed, complains that such probability has not been previously evaluated. *Id.* at 6-7. While there was and is no requirement for the NRC to evaluate the probability of such an accident, BCOC has not shown that there is a significant increase in the probability of a criticality accident in the Harris SFP, and does not raise an adequate basis for its contention.[14]

In addition, as set forth in the "NRC Staff Brief and Summary of Relevant Facts, Data and Arguments Upon Which the Staff Proposes To Rely At Oral Argument On Technical Contentions 2 and 3," January 4, 2000 (NRC Brief), BCOC is not qualified to analyze

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[14] BCOC adopts and incorporates in its Request for Admission its Summary and Appendix C to its "Detailed Summary of Facts, Data and Arguments and Sworn Submission On Which Orange County Intends To Rely At Oral Argument to Demonstrate the Existence of a Genuine and Substantial Dispute of Fact With the Licensee Regarding the Proposed Expansion of Spent Fuel Storage Capacity at the Harris Nuclear Power Plant With Respect to Criticality Prevention Issues," dated January 4, 2000 ("BCOC Subpart K Summary"). BCOC, however, does not identify any specific pages in its Subpart K Summary (out of 48 pages, exclusive of exhibits) or Appendix C thereto (out of 13 pages, exclusive of exhibits) as containing the information needed to establish a basis for its contention.

The Commission does not expect its adjudicatory boards, unaided by the parties, to sift through the parties' pleadings to uncover and resolve arguments not advanced by the litigants themselves. *Zion*, CLI-99-4, 49 NRC at 194. Rather, the burden of setting forth a clear and coherent argument is on the proponent of a contention. *Cf. id.* (ruling on intervention). BCOC purports to address the probability of criticality in Appendix C to its Subpart K Summary, but in no way connects any of the facts asserted therein to Harris.

criticality, and cannot establish that criticality could in fact occur at Harris, as asserted in Basis 2 for Technical Contention 2.[15] NRC Brief at 16-18. Moreover, the Staff and CP&L have demonstrated in this proceeding that criticality could not occur in the Harris SFP as claimed by BCOC. *Id.* at 28-31. For the reasons set forth above, BCOC has not provided any basis with respect to criticality for concluding that the proposed action will have a significant effect on the human environment, and there is no basis to contend that the Staff must prepare an EIS in connection with the proposed amendment because of criticality concerns.

Basis F-3 contends that the increased inventory and management of the spent fuel pools at Harris increase the opportunity for sabotage of the pools, and that the EA is inadequate because NRC has not considered new information regarding sabotage risks. The information provided does not support the contention.

Specifically, BCOC contends that the occurrence of a handful of terrorist events around the globe over the last seventeen years demonstrates that sabotage is a "reasonably foreseeable and significant threat" that must be addressed in an EIS. Request for Admission at 14. BCOC references Dr. Thompson's report, which relies chiefly on a 1996 book, a 1998 magazine article, and a February, 1999 newspaper article. Thompson Rep. at B-3-4 n. 5-7.

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[15] BCOC asserts that the GEIS is outdated and its findings are no longer applicable because new information shows the risks of a severe spent fuel accident during high-density fuel storage are significant. Request for Admission at 6-7. BCOC, while claiming that criticality can occur, does not analyze the nature of any criticality that might assertedly occur in the SFP, and does not establish that the consequences of any such criticality are significantly greater than the consequences of criticality considered in the GEIS. *See* GEIS, Section 4.2.3.4, at 4-19. BCOC's claim that the findings of the GEIS are no longer applicable with respect to criticality is devoid of support.

The first two references were available for use by BCOC well before the end of the time for timely contentions. Thus, any information drawn from these references on which Thompson bases his conclusions is not "new information" sufficient to support a good cause finding. *See Yankee Atomic Elec. Co.* (Yankee Nuclear Power Station), LBP-96-15, 44 NRC 8, 26 (1996)("Generally, a 'good cause' finding based on 'new information ' can be resolved by a straightforward inquiry into when the information at issue was available to the petitioner.") Moreover, the 1998 article, "Catastrophic Terrorism: Tackling the New Danger," discusses terrorism in general, with no specific mention of U.S. nuclear facilities or radiological sabotage. The 1999 article by Scott Allen, "NRC to Cut Mock Raids on Atom Plants," Boston Globe, Feb. 25, 1999, at A6, discusses NRC's plans to change its program that tests nuclear plant readiness for terrorists. The article does not discuss the Harris facility. To the extent that either of these articles provides "new information," BCOC fails to show a nexus between the contents of the Thompson Report and this information, on which it relies, and the Harris facility.

BCOC also attempts to distinguish *Limerick Ecology Action v NRC*, 869 F.2d 719 (3d Cir. 1989). The decision upheld NRC's refusal to accept for litigation in an operating license proceeding a contention regarding sabotage on the basis of the impossibility of modeling the risks of sabotage. Id. at 741-42. The examples provided by BCOC fail to demonstrate a specific nexus between these events and the Harris Application at issue here.

BCOC also contends that, although the risk of sabotage is not easily quantifiable, the NRC should address it in an EIS. The Thompson report agrees that the risk of sabotage is "less susceptible to probabilistic analysis" than other types of risk. Thompson Rep. at B-3.

As noted above, BCOC provides a laundry list of past sabotage events which have no relation to the Harris facility. The Thompson report suggests a few possible scenarios - a "sabotage event that leads to direct leakage from the pools;" "siphoning of water from the pools through . . . malice." Id. at C-1, C-5. These statements stand alone, and are not accompanied by any "statement of the alleged facts or expert opinion which supports the contention . . . together with 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." 10 C.F.R. § 2.714(b)(2)(ii). Therefore, the contention fails to comply with the requirements and must be dismissed. Moreover, a contention such as this one, that provides a postulated accident scenario - sabotage - must describe a causative mechanism for the accident. *See Vermont Yankee*, ALAB-919, 30 NRC at 44. Dr. Thompson's vague allusions to "sabotage events" are insufficient to meet this standard. For this contention to succeed, BCOC should have advanced "some method or theory by which the NRC could have entered into a meaningful analysis of the risk of sabotage despite its asserted inability to quantify the risk. *Limerick*, 869 F.2d at 744.

Finally, BCOC contends that a consideration of the environmental impacts of sabotage should incorporate severe accident mitigation design alternatives (SAMDAs) which could mitigate the impacts of sabotage. BCOC appears to state that wet storage options involve severe accident risks, (such as draining the spent fuel pool) , and dry storage options do not. The notion of a loss of water inventory in the spent fuel is remote and speculative, and BCOC has not provided a basis to support why this scenario needs to be considered for the purposes of NEPA. In 1985, in issuing its Severe Accident Policy Statement, 50 Fed.

Reg. 32138, 32144, the NRC concluded that '[o]perating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new safety information arises to question whether there is adequate assurance of no undue risk to public health and safety." This proposed contention offers no such significant new information. In any event, Harris is required, like all power plants, to maintain a safeguards contingency plan that includes plans for dealing with "threats, thefts, and radiological sabotage." *See* 10 C.F.R. § 50.34(d).

In sum, BCOC's contention that the Staff should prepare an EIS to discuss the risks of sabotage introduced by pools C and D is without support and should be dismissed.

> <u>CONTENTION EC-2</u>: The EIS is deficient because it fails to acknowledge or evaluate the significant environmental risk posed by the operation of pools A, B, C, and D.

<u>Basis:</u> The NRC is required by law to evaluate the cumulative impacts of operation of pools C and D, in conjunction with the impacts of current operation, including operation of pools A and B. New information, developed since the publication of the 1979 GEIS, shows that it "constitutes an inadequate basis for drawing any conclusions about the environmental impacts of operating pools A & B." Request for Admission at 17. The new information shows that there is a significant risk that a degraded-core reactor accident will lead to a SPF accident. Therefore, the NRC is required to perform an integrated risk evaluation of all pools, including how the pool loading pattern would influence accident risk and how the potential for an accident at one pool could affect the development of an accident at another pool. Request for Admission at 17-18.

Staff Response to EC-2. This contention must be dismissed because it does not state an adequate basis for admission. The basis refers to "new information," yet nowhere in the basis is the new information specified. A reference to EC-1, Section E is made to support the claim of new information, but EC-1, Section E contains no such new information. In fact, as demonstrated elsewhere in this brief, BCOC has offered *no* new information regarding the probability that a degraded core accident would lead to an SFP accident, just the unsupported opinion and conclusions of its consultant. An expert opinion must provide a sufficient basis for the conclusions reached and no such basis has been demonstrated here. BCOC has not demonstrated that there are any cumulative effects to be analyzed.

> CONTENTION EC-3: 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.

Basis. BCOC bases this contention on the assertion that the purpose of the proposed expansion is to store fuel not only from Shearon Harris, but also from Brunswick and Robinson. Despite the fact that CP&L has a dry storage facility at Robinson and has applied for an ISFSI license for Brunswick, which represent "viable alernative[s] to high-density storage in pools C and D," BCOC contends that the Staff "should be required to thoroughly examine the alternative of dry storage in an EIS." Request for Admission at 19.

Staff Response to Contention EC-3. This contention should be dismissed because the proposed action does not involve the authorization to receive spent fuel from Brunswick and Robinson. The operating license issued for Harris authorized the receipt of spent fuel from Robinson and Brunswick and the receipt of such fuel was acknowledged in the 1983 SER at 9-6. Since Harris is already authorized to receive spent fuel from Brunswick and

Robinson, and is not seeking an amendment to that license condition, this contention is not relevant to this proceeding. *See Virginia Elec. Co.* (North Anna Power Station, Units 1 & 2), ALAB-790, 20 NRC 1450, 1453-54 (1984) (ruling that an amendment to permit the receipt and storage of spent fuel at the North Anna facility from the Surry facility has no bearing on a separate amendment approving the expansion of the spent fuel pool at the North Anna facility).

> CONTENTION EC-4: 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). Request for Admission at 20.

Basis: BCOC contends that special circumstances exist warranting a discretionary EIS because the proposed action involves unresolved conflicts concerning alternative use of available resources within the meaning of section 102(2)(E) of NEPA. *Id.* According to BCOC, these conflicts arise from the fact that CP&L intends to store spent fuel from three different reactors: Harris, Brunswick and Robinson. *Id.*

Staff Response to Contention EC-4. The question of whether the Staff should prepare an EIS pursuant to 10 C.F.R. §§ 51.20(b)(14) and 51.22(b) is a matter of Staff discretion. The Board does not have the authority to direct the Staff to prepare an EIS as a matter of discretion. *See Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plants, Units 1, 2, 3 & 4), CLI-80-12, 11 NRC 514, 516-17 (1980) ("[T]he Boards do not direct the staff in performance of their administrative functions.").

None of BCOC's claims demonstrate that an EIS should be prepared as a matter of discretion. As characterized by BCOC, the unresolved conflicts arise from the storage of spent fuel from the Brunswick and Robinson facilities. The storage of spent fuel from Robinson and Brunswick is already authorized by the Harris license. Thus, the consideration of the environmental impacts of the storage of spent fuel from Robinson and Brunswick is not warranted. *See North Anna,* ALAB-790, 20 NRC at 1453-54.

BCOC also asserts another reason to prepare an EIS is that the Licensee's proposal appears to be in conflict with the Commission's Waste Confidence decision. *See* Request for Admission at 22-23, *citing* 10 C.F.R. § 51.23. According to 10 C.F.R. § 51.23, the Commission stated its belief that there is reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century. 10 C.F.R. § 51.23(a). CP&L, however, stated in its application that DOE spent fuel storage facilities are not available and are not expected to be available for the foreseeable future. Request for Admission at 22, *citing* Licensee Application, Enclosure 1 at 1. Thus, BCOC contends, the license amendment application is in conflict with 10 C.F.R. § 51.23. *Id.* This concern also does not constitute a special circumstance warranting the preparation of a discretionary EIS. According to the Licensee Application, CP&L anticipates a need for an expansion in spent fuel storage capacity by the year 2000. 10 C.F.R. § 51.23 provides that there is reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century, but not necessarily by the year 2000. *See* 10 C.F.R. § 51.23. Thus, there is no conflict between the basis of CP&L's proposal and the Commission's regulation. In any event, 10 C.F.R. § 51.23 only relates to the environmental impacts of spent fuel storage beyond the operating term of a reactor. BCOC, thus, fails to demonstrate that special circumstances exist warranting a discretionary EIS.

III. CONCLUSION

Based upon the foregoing, the Staff submits that the requirement of supporting a contention with a "statement of the alleged facts or expert opinion which supports the contention . . . together with 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," (10 C.F.R. § 2.714(b)(2)(ii)) has not been met as to any of the four contentions offered by BCOC and the bases are not admissible in support of contentions. BCOC has not demonstrated that there are any genuine and substantial disputes of material fact as to any aspect of the contentions and there is no issue raised in the contentions which require the introduction of evidence in an adjudicatory proceeding for resolution.

Respectfully submitted,

for

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Robert M. Weisman
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Dated at Rockville, Maryland
this 3rd day of March 2000.

EXHIBIT "A"

ERI/NRC 95-103

TECHNICAL EVALUATION REPORT OF THE SHEARON HARRIS INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

Final Report

May 1995

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P.O. Box 2034
Rockville, Maryland 20847

Prepared for:
SCIENTECH, Inc.
Rockville, Maryland

Under Contract NRC-04-91-068
With the United States Nuclear Regulatory Commission
Washington, D.C. 20555

Energy Research, Inc.

plant does not permit communication between the cavity and the upper compartment. Therefore, the submittal assumes that only a small fraction of the core debris ejected at high pressure during vessel breach will be transported to the lower compartment of the containment and participate in direct containment heating. Secondly, the Shearon Harris containment was determined to have a large capacity (i.e., the median containment failure pressure is calculated to be 153 psig).

The small probability of late containment failure is attributed to the following two reasons. First, the containment has a large cavity floor area, and therefore the submittal assumes that there is a high conditional probability of coolability of debris on the cavity floor by an overlying pool of water. Secondly, the concrete type in the Shearon Harris plant is a quartz-based aggregate, which is similar to Basaltic concrete. The generation of non-condensible gases were found to be very small for this type of concrete, and hence the conditional probability of late overpressure failure is calculated to be low.

The submittal indicates a high conditional probability of releases due to induced steam generator tube rupture events. The EOPs for the Shearon Harris plant require the operators to restart the Reactor Coolant Pumps (RCPs) if they are available, when there is inadequate core cooling.

Table E.1 Containment Failure as a Percentage of Total CDF: Comparison With Other PRA Studies

Containment Failure Mode	Shearon Harris IPE	Surry NUREG-1150	Zion NUREG-1150
Containment Failure with In-Vessel Recovery (Prior to Vessel Breach)	3.2	-**	-**
Early Failure	0.25	0.7	0.5
Late Failure	1.0	5.9	24.0
Very Late Failure	3.6	NA+	NA+
Bypass (V)	0.7	7.6	0.2
Bypass (SGTR)	6.5	4.6	0.3
Isolation Failure	0.3	NA++	1.0
Intact	84.5	81.2	73.0
Core Damage Frequency, yr^{-1}	$7x10^{-5}$*	$4.1x10^{-5}$	$6.2x10^{-5}$

* Includes Flooding

** Not Applicable

++ Included as a Part of Early Containment Failure

\+ Included as a Part of Late Containment Failure review of the results from severe accident analyses did not identify any volumes inside the containment where a sufficient buildup of hydrogen could lead to Deflagration-to-Detonation Transition (DDT). However, the licensee stated that a containment walkdown has not been performed to identify passages conducive to the occurrence of DDT. In summary, all the CPI recommendations are addressed by the licensee.

E.5 Vulnerabilities and Plant Improvements

The submittal does not define "vulnerability", particularly as related to containment analyses. However, it should be noted that a few initiators, particularly the loss of offsite power and the small break LOCA dominate the CDF profile. In addition, it should also be noted that the EOP requiring the operators to restart the RCP to provide additional cooling during the course of severe accidents was found to have important consequences. In the submittal, this operator action is shown to lead to induced SGTR and thereby large radiological releases. In response to an NRC question on the induced SGTR (due to reactor coolant pump restart), the licensee stated that this failure mode of steam generator tubes is not unique to the Shearon Harris plant, and that the issue is being addressed by the Westinghouse Owners Group. It should be noted that this potential failure mode has not been identified by other IPEs reviewed by ERI, for other Westinghouse PWRs.

A multi-disciplinary utility team was formed to review the IPE results and suggest plant modifications and improvements. Although this team made several suggestions for improvements to the plant, no improvements based on the containment analyses were considered necessary.

E.6 Observations

The assessment of this review is that the Shearon Harris IPE submittal documentation, and the responses to the NRC review team questions, contains substantial *Back-End* information regarding the severe accident vulnerability issues for the Shearon Harris plant.

The following are the major findings of the Shearon Harris IPE submittal:

- The Shearon Harris submittal shows that the overall CDF of 7×10^{-5} per reactor year is not dominated by any single initiating event. Small LOCAs and loss of offsite power sequences contribute to more than 60% of the total CDF.

- The containment analyses indicate that there is a 15% conditional probability of releases, and 85% conditional probability of intact containment.

- Containment features such as large cavity floor area, limited communication between the cavity and the containment, and the larger calculated containment capacity, all contribute to the low conditional probability of containment failure.

2.3.2 Dominant Contributors to Containment Failure

The containment failure modes and timings for various accident sequences are provided in Section 4.6.3 and summarized in Section 4.8 of the submittal. Table 6 of this review shows a comparison of the conditional probabilities of the various containment failure modes of the Shearon Harris IPE submittal with the Surry and Zion NUREG-1150 results. All comparisons are made for internal initiating events only.

Table 6 Containment Failure as a Percentage of Total CDF: Comparison With Other PRA Studies

Containment Failure Mode	Shearon Harris IPE	Surry NUREG-1150	Zion NUREG-1150
Very Early Failure with In-Vessel Recovery (Prior to Vessel Breach)	3.2	-**	-**
Early Failure	0.25	0.7	0.5
Late Failure	1.0	5.9	24.0
Very Late Failure	3.6	NA+	NA+
Bypass (V)	0.7	7.6	0.2
Bypass (SGTR)	6.5	4.6	0.3
Isolation Failure	0.3	NA++	1.0
Intact	84.5	81.2	73.0
Core Damage Frequency, yr^{-1}	$7x10^{-5}$*	$4.1x10^{-5}$	$6.2x10^{-5}$

* Includes Flooding

** Not Applicable

++ Included as a Part of Early Containment Failure

\+ Included as a Part of Late Containment Failure

The Shearon Harris core damage frequency for internal events is slightly larger than that calculated by NUREG-1150 for Surry and Zion [4,5]. The conditional probability of early containment failure (due to overpressurization) in the Shearon Harris plant is 0.25% and is considerably less than that calculated for the Zion and Surry plants. This is primarily due to the treatment of the phenomena that threaten the containment integrity at vessel breach, such as DCH, steam explosions, etc. The RCS is assumed to be depressurized at vessel breach for a large fraction of accident sequences in the Shearon Harris IPE. However, the submittal

EXHIBIT “B”

NUREG/CR-0649
SAND77-1371
R-3

SPENT FUEL HEATUP FOLLOWING LOSS OF WATER DURING STORAGE

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Date Published: March 1979

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5. OTHER CONSIDERATIONS

5.1 Effect of Incomplete Drainage

Many spent fuel holder designs provide only a single inlet hole for convective flow through each fuel element, located in the baseplate or near the bottom of the holder. If there is a complete pool drainage, the air must circulate down and under the fuel elements before passing through the baseplate inlet hole into the fuel assembly. An incomplete drainage could block this flow and reduce the effectiveness of natural convective cooling. Open frame configurations are, of course, exempt from this possibility because the flow does not have to pass through an inlet hole in order to gain proximity to the fuel element.

A detailed analysis of spent fuel heatup in the event of an incomplete drainage has not been undertaken. However, an approximate analysis has been performed to estimate the amount of aggravation that might occur if the water ceased to drain after exposing all but the bottom portion of the fuel elements. The analysis is included in Appendix B and is based, among other things, upon upper and lower bound estimates of the thermal radiation absorbed by the water from the hot fuel rods above. The temperature distribution along the rods is prescribed in this analysis according to estimates made of the likely distribution that would occur just prior to the onset of self-sustaining clad oxidation. The amount of heat produced above the water level is then determined together with the amount that could be removed by various mechanisms, including water boiling (latent heat), convection to the steam produced by boiling (sensible heat), radiation to the building, and convection to the air. If the heat removal rate is determined to be larger than the rate of production, then the configuration is coolable; if the heat removal rate is smaller than the rate of production, overheating resulting in clad rupture or melting will occur.

The results for a 1-year decay time are presented in Table VIII. Consider first the case where the drainage uncovers the upper 80 percent of the fuel rods, leaving the lower 20 percent still covered (third column). The heat transferred to the remaining water by decay from the immersed portions and by radiation from above is 3.6 - 4.9 KW per assembly (line 2c). This implies that about an hour might be required to raise the water temperature to boiling (assuming all the assemblies produce the same decay heat) and that the water recession rate following the inception of boiling will be about 10 cm/h (lines 3 and 4). Meanwhile, the decay heat produced above the water line is about 4.5 KW per assembly (line 5), and the capability for removing heat as the clad temperatures approach the lower limit of self-sustaining oxidation is 5.7 - 8.7 KW per assembly (line 6e). Since the heat removal capability exceeds the heat production (line 7), the geometry is temporarily coolable.

If, however, the drainage were to uncover the whole length of the rods but still to constrict the flow, either by blocking the baseplate holes or by not allowing enough space for unrestricted flow in the base region, then the heat production would exceed the heat removal capability (line 7, first column) and the clad would overheat. The same situation would eventually occur if, rather than immediately draining to this position, the water were to drain part way down the rods and then boil off down to the baseplates over a period of time. Table VIII indicates that there is a good chance of overheating, in

Table VIII.

Estimates of Heat Removal Capability in an Incompletely Drained Pool, One Year Decay Time*

1. Normalized water level (z_w/L)	0.0	0.1	0.2
2. Heat transferred to water, per assembly (KW):			
a. by decay heat	0.0	0.2	0.6
b. by thermal radiation from above	0.3 - 1.3	1.2 - 2.6	3.0 - 4.3
c. total	0.3 - 1.3	1.4 - 2.8	3.6 - 4.9
3. Time to start boiling (hours)	1.0 - 4.3	0.9 - 1.8	0.7 - 1.0
4. Water surface recession rate (cm/hr)	0.7 - 3.2	3.5 - 7.0	9.0 -12.2
5. Decay heat produced by spent fuel above water level, per assembly (KW)	5.1	4.9	4.5
6. Removal of heat produced by spent fuel above water level, per assembly (KW):			
a. by radiation to water	0.3 - 1.3	1.2 - 2.6	3.0 - 4.3
b. by radiation to building	0.0 - 0.9	0.0 - 0.9	0.0 - 0.9
c. by transfer to water vapor	0.2 - 0.8	0.9 - 1.8	2.3 - 3.1
d. by transfer to air	0.4	0.4	0.4
e. total	0.9 - 3.4	2.5 - 5.7	5.7 - 8.7
7. Heat removal surplus (deficit) per assembly (KW), line 6e minus line 5.	(4.2)-(1.7)	(2.4)-0.8	1.2 - 4.2

* PWR spent fuel in cylindrical baskets. One year decay time assumed, uniformly throughout pool. Numerical ranges (e.g., 0.3 - 1.3) give lower and upper-bound estimates. See Appendix B.

fact, if the water were to recede below the level where the lower 10% of the rods is still immersed.

A comparison of the peak clad temperature rise versus time for PWR spent fuel with a 1-year minimum decay time in a well-ventilated room is shown in Figure 26. The temperature rise corresponding to an incomplete drainage down to the bottom of the rods, calculated by utilizing the lower-bound radiation estimate, is compared with previous cases for a complete drainage with varying baseplate hole sizes. The clad oxidation effect has not been calculated for the case of incomplete drainage (blocked inlets), because it is believed to be substantially reduced by the unavailability of oxygen within the assembly. Clearly, a 1-year minimum decay time is not sufficient to preclude overheating for this case.

The approximate method used for bracketing the thermal radiation downward to the water and upward to the building is not considered to be precise enough to allow prediction of the minimum allowable decay time in the event of an incomplete drainage. This problem could be approached by formulating a detailed thermal radiation model to calculate shape factors and include the shadowing of radiating surfaces by fuel rods and tie plates. By incorporating this radiation capability into the overall heat transfer models described in Sections 3.3 and 3.4, a credible prediction of the minimum allowable decay time could be obtained. No attempt to do this, however, has been made.

It is clear, however, that an incomplete drainage can potentially cause a more severe heatup problem than a complete drainage, if the residual water level remains near the baseplates. From a practical point of view, it might be possible to make provisions for either completing the drainage or refilling the pool, if this should happen. However, it would

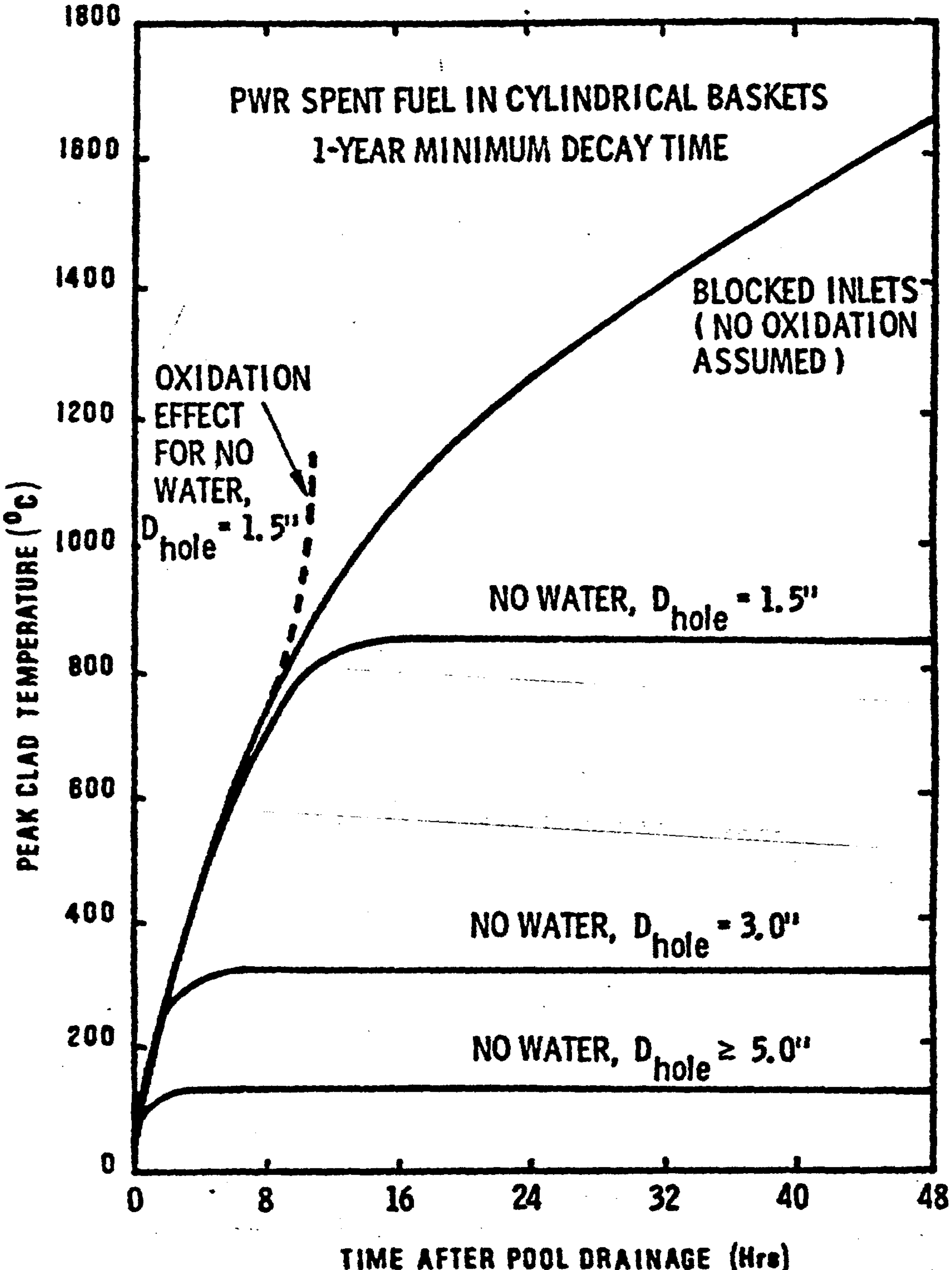

Figure 26. Estimated Heatup of PWR Spent Fuel With Residual Water Sufficient to Block Flow Inlets, Well-Ventilated Room

seem that the special problems associated with an incomplete drainage could best be circumvented by modifying the spent fuel holders to include inlet holes at various elevations along the vertical, rather than just at the baseplate level. According to the predictions, these inlet holes would only be required for the bottom 20 percent of the fuel rod length if the spent fuel were at least a year old. With these additional inlets, the beneficial effect of natural convection would not be cancelled by an incomplete drainage.

5.2 Effect of Surface Crud

Iron oxides are known to deposit upon the outside of the fuel pins during normal operation of the reactor, and these deposits are likely to remain on the fuel pins during storage of the spent fuel. Typically, the iron oxide crud buildup on BWR fuel pins is on the order of 25 to 100 microns and in the form of Fe_2O_3, whereas the buildup on PWR pins is on the order of only 1 to 5 microns and in the form of Fe_3O_4.[16] A calculation was made to determine whether a 100 micron Fe_2O_3 coating on the BWR fuel pins would affect the heatup of these pins during a pool drainage accident, and it was found that the overall effect on the fuel pin temperature was less than one degree.

The question was also raised as to whether some of the crud, which would be contaminated, could be levitated by the air flows produced by natural convection after a pool drainage and thereby produce a health hazard. An analysis of the weight and drag characteristics of iron oxide particles revealed that a BWR fuel assembly having a decay time of 90 days prior to loss of water can produce upward air currents sufficient to levitate a 200-micron sized particle, whereas an assembly allowed to decay for 250 days can levitate a 175-micron sized particle. Since any spallation of the crud would produce particles of roughly the same size as the thickness of the

6. CONCLUSIONS

An analysis of spent-fuel heatup following drainage of the storage pool has been completed, and the following conclusions have been reached:

Well-Ventilated Rooms

1. Considering a complete pool drainage, the minimum allowable decay time for PWR spent fuel in a well-ventilated room varies from a best value of about 5 days, for open-frame storage configurations, to a worst value of about 700 days, for high-density closed-frame configurations with wall-to-wall spent fuel placement. Other storage configurations fall between these limits. The minimum allowable decay time is defined as the lower limit of safe decay times, such that shorter decay times would produce local clad failures due to rupture or melting.
2. The minimum allowable decay time for BWR spent fuel in a well-ventilated room varies from a best value of 5 days to a worst value of 150 days for the cases considered. A high-density storage rack design for BWRs would result in a somewhat higher value of the allowable decay time than presented here, but not as high as for PWR spent fuel.
3. The allowable decay times can be reduced significantly by widening baseplate holes, opening flow paths between holders, removing BWR channels, and avoiding wall-to-wall storage. Decay times as low as 80 days for the high density racks and 20 days for other

racks could in principle be accommodated with these design modifications at no expense in packing density.

4. The differences between fuel assembly designs are small, i.e., a 17 x 17 PWR pin array and a 15 x 15 PWR pin array produce similar results, as do an 8 x 8 BWR pin array and a 7 x 7 BWR pin array. The effect of surface crud on the fuel pins is also insignificant.

Inadequately Ventilated Rooms

5. Current forced air ventilation systems in typical PWR auxiliary buildings may provide insufficient ventilation to remove the decay heat produced in the spent fuel pool after a complete pool drainage. Consequently, overheating due to inadequate ventilation may occur. Adequate ventilation could be provided by passive methods that utilize a chimney effect.
6. Ventilation systems in typical BWR spent fuel pools inside the reactor containment building are adequate to remove most of the decay heat, owing to the large size of the containment building.
7. Additional ventilation provisions for typical away-from-reactor facilities (750 MTU capacity) will be unnecessary if the spent fuel is sufficiently aged. Minimum decay times of between 2 and 4 years, depending on the storage configuration, are sufficient to prevent overheating in AFR storage pools with inadequate or inoperative ventilation because of the fairly substantial size of the room, the presence of heat sinks, and the capacity of the sheet metal walls to reject heat through thermal radiation to the outside. Shorter decay times can be accomodated by providing additional passive ventilation.

Incomplete Drainage

8. For many spent fuel holder designs where the air must circulate under the fuel elements and pass through a

baseplate hole to enter the elements, a nearly complete drainage can be more severe than a complete drainage. For 1-year-old spent fuel, coolability can be maintained by the process of water boiling and convection of heat to the steam, as long as the lower 20 percent of the fuel rods remains covered by water. If the water drains or boils off to a lower level, but not sufficiently low to open the baseplate passages to air flow, then the removal of heat associated with water boiling, steam convection, and air convection will all be impaired. These circumstances can lead to an increased tendency to overheat.

9. The potentially adverse effects of an incomplete drainage can be counteracted by drilling air inlet holes at various elevations in the lower part of the holders. This will permit air flows to circulate when the water level drops beneath the location of the uppermost inlet holes.

Emergency Water Spray

10. For those cases where overheating is a concern, coolability in a drained spent fuel pool can be maintained indefinitely by providing a water spray. Spray volumes on the order of 100 gal/min and less appear to be sufficient for all the cases considered. The gamma dose rate to a person entering within 50 feet of the edge of the pool to set up a fire hose is about 200 Rem/hr.

EXHIBIT “C”

NUREG-1038

Safety Evaluation Report

related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2

Docket Nos. STN 50-400 and STN 50-401

Carolina Power and Light Company
North Carolina Eastern Municipal Power Agency

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

November 1983

9 AUXILIARY SYSTEMS

FSAR Chapter 9, "Auxiliary Systems," has been reviewed in accordance with the SRP (NUREG-0800). The staff has reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, and fuel storage.

The auxiliary systems necessary for safe reactor operation or shutdown include the essential service water system (ESWS), the component cooling water system (CCWS), the ultimate heat sink (UHS), the condensate storage facility, the essential services chilled water system, the control room area ventilation systems, and the engineered safety feature ventilation system.

The auxiliary systems necessary to ensure the safety of the fuel storage facility include new fuel storage, spent fuel storage, the spent fuel pool cooling and cleanup system, fuel-handling systems, and the spent fuel pool area ventilation system.

The staff has also reviewed other auxiliary systems to verify that their failure will not prevent safe shutdown of the plant or result in unacceptable release of radioactivity to the environment. These systems include the nonessential service water system, the demineralized water makeup system, potable and sanitary water system, the nonessential services chilled water system, the waste processing building cooling water system, the compressed air systems, the equipment and floor drainage systems, the turbine building area ventilation system, and the control rod drive mechanism ventilation system. This review also included nonessential portions of the essential systems discussed above.

9.1 <u>Fuel Storage and Handling</u>

9.1.1 New Fuel Storage

The new fuel storage facility was reviewed in accordance with SRP 9.1.1.

The acceptance criteria for the new fuel storage facility include meeting ANS 57.1, "Design Requirements for Light-Water Reactor Fuel Handling System," and ANS 57.3, "Design Requirements for New LWR Storage Facilities." The SRP guidelines were used in lieu of ANS 57.1 and ANS 57.3.

The new fuel storage facility is located in the fuel-handling building. Although the new fuel is stored dry, the new fuel storage facility is designed to also be used as optional wet storage space for spent PWR fuel. The storage facility consists of two new fuel pools, one for each unit. The pools are interconnected by means of a transfer canal whose length is that of the fuel-handling building. However, the two new fuel pools are normally isolated by removable gates designed to seismic Category I standards.

Each new fuel pool can store 580 PWR fuel assemblies; this constitutes enough fuel for more than three fuel cores. The new fuel is stored in rack modules of several designs that can be removed and installed under water. There are separate new fuel pools for each unit. Nevertheless, the new fuel storage system is designed so that new fuel for either unit may be stored in either new fuel pool, space permitting. However, no failure resulting from sharing of the storage area prevents safe shutdown of either or both units. Therefore, the requirements of GDC 5 are satisfied.

The fuel-handling building, which houses the facility, and the storage racks and pools are designed to seismic Category I criteria. This building is also designed against flooding and tornado missiles (see Sections 3.4.1 and 3.5.2 of this SER). Thus, GDC 2 and 4 and RG 1.29, Position C.1, are satisfied.

The new fuel storage facility is not located in the vicinity of any moderate- or high-energy lines or rotating machinery. Separation from such potential missile sources protects the new fuel from internally generated missiles and the effects of pipe breaks (see Sections 3.5.1.1 and 3.6.1 of this SER).

Accidental damage to the new fuel would release relatively minor amounts of radioactivity that would be accommodated by the spent fuel pool area ventilation system. Thus, GDC 61 is satisfied.

The applicant stated that the new fuel storage racks are designed so that with fuel of the highest anticipated enrichment and with the pool flooded with unborated water, K_{eff} will still be 0.95 or less. If the unborated water is replaced by moderators such as foam or water mist, K_{eff} will still be 0.95 or less. The staff is making an independent evaluation of new fuel reactivity under the most adverse conditions to ensure that the new fuel storage facility complies with GDC 62 with regard to criticality as a result of storage of new fuel assemblies.

The storage racks can withstand an uplift force equal to the maximum uplift capability of the spent fuel bridge crane and are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. Thus, GDC 62 is satisfied.

Based on its review, the staff concludes that the new fuel storage facility is in conformance with GDC 2, 4, 5, and 61 as they relate to new fuel protection against natural phenomena, missiles, shared functions, and radiation protection and prevention of criticality, and with 1.29, Position C.1, as it relates to seismic classification. The staff is making an independent evaluation as to whether the design of the new fuel storage facility complies with the requirements of GDC 62. Pending completion of this evaluation, the staff concludes that the design of the new fuel storage facility meets SRP 9.1.1.

9.1.2 Spent Fuel Storage

The spent fuel storage facility was reviewed in accordance with SRP 9.1.2.

The acceptance criteria for the spent fuel storage facility include meeting various portions of the guidelines of ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." The guidelines contained in the SRP were used in lieu of ANS 57.2. The acceptance criteria also include RG 1.115, "Protection Against Low-Trajectory Turbine Missiles." Tu-bine missiles are evaluated in Section 3.5.1.3 of this SER.

A spent fuel storage facility is provided for each unit. These pools are interconnected by the main fuel transfer canal, but they are normally isolated by removable gates designed to seismic Category I standards. The maximum storage capacity of the two spent fuel pools is 3024 PWR assemblies, which is more than 19 full PWR cores. Fuel may be stored in a combination of 6 x 10, 6 x 8, and 7 x 7 PWR rack modules. The applicant stated that rack rearrangement would have no effect on stored fuel criticality. The spent fuel pools may also be used for the storage of BWR fuel; the 7 x 7 PWR fuel storage modules may be interchanged with 11 x 11 BWR fuel storage modules because both cover the same floor area. The PWR racks have a center-to-center distance of 10.5 inches between cells, while the BWR racks have a center-to-center distance of 6.25 inches. PWR fuel assemblies cannot be inserted into BWR rack modules, but BWR fuel assemblies can be inserted into PWR rack modules. Introduction of a BWR fuel assembly into a PWR rack module, however, will result in a subcritical array of fuel assemblies with $K_{eff} \leq 0.95$. The applicant has provided sufficient information regarding fuel enrichment, geometry of spent fuel and racks in pool, and calculational methodology to permit the staff to make an independent evaluation of spent fuel pool reactivity under the most adverse conditions. The staff has reviewed the information provided and concludes that $K_{eff} \leq 0.95$ under the most adverse conditions. Therefore, the staff concludes that the spent fuel facility complies with a GDC 62.

The structure housing the spent fuel storage facility (the fuel-handling building) is designed to seismic Category I criteria, as are the storage racks, pool liners, gates, canals, and storage pools. The building is also designed against flooding and tornado missiles (see Sections 3.4.1 and 3.5.2 of this SER). The staff concludes that GDC 2 and RGs 1.13, Position C.3; 1.29, Positions C.1 and C.2; and 1.117, Positions C.1 through C.3, are satisfied for the spent fuel storage facility.

The spent fuel storage facility is not in the vicinity of any high-energy lines or rotating machinery. Therefore, physical protection by means of separation is utilized to protect the spent fuel from internally generated missiles and the effects of pipe breaks (see Sections 3.5.1.1 and 3.6.1 of this SER). Thus, GDC 4 and RG 1.13, Position C.3, are satisfied.

The shared portion of the facility has sufficient redundancy of services and is of seismic Category I, Quality Group C design, so that an accident in one unit with loss of offsite power will not impair its ability to safely store the spent fuel. This satisfies GDC 5.

The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the spent fuel bridge crane. In addition, loads greater than a fuel assembly are not carried over spent fuel, in compliance with the guidelines of Position C.5.c of RG 1.13 (see Sections 9.1.4 and 9.1.5 for discussion of protection of spent fuel in both new and spent fuel pools against damage from light and heavy loads).

The fuel-handling building has a ventilation system (the spent fuel pool area ventilation system) to limit the potential release of radioactivity in the event of an accident (see Section 9.4.2 of this SER for a discussion of the spent fuel pool area ventilation system) in accordance with Position C.4 of RG 1.13. In this way, the spent fuel storage facility complies with GDC 61.

Control room and local alarms are provided to alert the operator to high and low pool water level and high temperature in the fuel pool. The fuel-handling building has a radiation monitoring system. These features satisfy GDC 63.

The staff also has reviewed the compatibility and chemical stability of the materials of the spent fuel pool and storage racks wetted by the pool water. The pool liner, rack lattice structure, and fuel storage tubes are stainless steel. The pool contains oxygen-saturated demineralized water containing 2000 to 4000 ppm boron as boric acid.

In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00×10^{-5} in. in 100 years (Weeks, 1977), which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials. Provisions are incorporated to allow for the periodic inspection of spent fuel pool components.

The staff, therefore, concludes that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate based on test data and actual service experience in operating reactors. The staff also concludes that the selection of appropriate materials by the applicant meets GDC 61 by having a capability to permit appropriate periodic inspection and testing of components, and GDC 62 by preventing criticality by maintaining structural integrity of components.

Based on its review, the staff concludes that the spent fuel storage facility is in conformance with GDC 2, 4, 5, 61, 62, and 63 as they relate to protection of spent fuel against natural phenomena, missiles, environmental effects, the facility's shared functions, radiation protection, periodic inspecting and testing of components, prevention of criticality, and performance monitoring, and with RG 1.13, Positions C.1, C.3 and C.4; 1.29, Positions C.1 and C.2; and 1.117, Positions C.1 through C.3, relating to the facility's design, seismic classification, and protection against tornado missiles. Therefore, the staff concludes that the spent fuel storage facility meets SRP 9.1.2.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system was reviewed in accordance with SRP 9.1.3.

The acceptance criteria for the cooling portion of the fuel pool cooling and cleanup system (FPCCS) include meeting RG 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," if the cooling portion of the system does not meet GDC 2. This does not apply to Shearon Harris because the fuel pool cooling portion of the FPCCS meets GDC 2, as discussed below.

EXHIBIT "D"

NUREG-0972

Final Environmental Statement

related to the operation of Shearon Harris Nuclear Power Plant, Units 1 and 2

Docket Nos. STN 50-400 and STN 50-401

Carolina Power and Light Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1983

SUMMARY AND CONCLUSIONS

This Final Environmental Statement was prepared by the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (the staff).

1. This action is administrative.

2. The proposed action is the issuance of operating licenses to Carolina Power and Light Company (applicant) for the operation of Shearon Harris Units 1 and 2 (NRC Docket Nos. 50-400 and 50-401), located in Wake and Chatham Counties, North Carolina, approximately 26 km (16 miles)* southwest of Raleigh, the state capital. The two units will employ three-loop, pressurized-water reactors (PWRs) to produce a rated 2785 MWt of heat generated in the core, which includes 10 MWt from the reactor coolant pumps, and which is converted to produce approximately 951 MW of electricity (gross dependable capacity). The plant employs a closed-cycle cooling system; the primary heat sink is the atmosphere, through a natural draft cooling tower for each unit. Makeup water for the cooling towers is drawn from a manmade reservoir.

3. The information in this statement represents the second assessment of the environmental impacts pursuant to the Commission's regulations as set forth in Title 10 of the Code of Federal Regulations Part 51 (10 CFR 51), which implements the requirements of the National Environmental Policy Act of 1969 (NEPA). After receiving, in September 1971, an application to construct Shearon Harris Units 1, 2, 3, and 4, the staff carried out a review of impacts that would occur during station construction and operation. That evaluation was issued as a Revised Final Environmental Statement-Construction Permit phase (RFES-CP) in March 1974. After this environmental review, a safety review, an evaluation by the Advisory Committee on Reactor Safeguards, and public hearings, the U.S. Nuclear Regulatory Commission issued Construction Permits Nos. CPPR-158, 159, 160, and 161 in January 1978. The applicant submitted an application for an operating license (OL) by letter dated June 26, 1980. The NRC conducted a predocketing acceptance review and determined that sufficient information was available to start detailed environmental and safety reviews. The FSAR was docketed on December 22, 1981. The applicant on December 18, 1981 informed the NRC that Units 3 and 4 had been cancelled, and on January 7, 1982 requested that Units 1 and 2 be considered concurrently for operating licenses.

*Throughout the text of this document, values are generally presented in both metric and English units. (Exceptions are sometimes made in areas where the accepted standard in the discipline is expressed in English units.) For the most part, measurements and calculations were origially made in English units and subsequently converted to metric. The number of significant figures given in a metric conversion is not meant to imply greater or lesser accuracy than that implied in the original English value.

DOCKETED
USNRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD '00 MAR -6 P3:58

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

NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney enters an appearance in the above-captioned matter. In accordance with § 2.713(b), 10 C.F.R., Part 2, the following information is provided:

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Name of Party:	NRC Staff

Respectfully submitted,

Brooke D. Poole

Brooke D. Poole
Counsel for NRC Staff

Dated at Rockville, Maryland
this 3rd day of March, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO INTERVENOR'S REQUEST FOR ADMISSION OF LATE-FILED ENVIRONMENTAL CONTENTIONS," and "NOTICE OF APPEARANCE" for Brooke D. Poole in above-captioned proceeding have been served on the following through deposit in the NRC's internal mail system, or by deposit in the NRC's internal mail system, with copies by electronic mail, as indicated by an asterisk, or by deposit in U.S. Postal Service as indicated by double asterisk, with copies by electronic mail as indicated this 3rd day of March, 2000:

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