RAS 2403UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 11/21/00

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY AT ORAL ARGUMENT ON ENVIRONMENTAL CONTENTION EC-6

I. INTRODUCTION

Pursuant to 10 C.F.R. § 2.1113, the Nuclear Regulatory Commission staff (Staff) hereby submits its written presentation summarizing all the facts, data and arguments of which the Staff is aware and on which the Staff intends to rely at oral argument, scheduled for December 7, 2000. For the reasons set forth below, the Staff submits that there is no genuine and substantial dispute of fact or law relating to the Board of Commissioners of Orange County's (BCOC) Environmental Contention EC-6. This written summary is supported by the affidavit of Gareth W. Parry, Stephen F. LaVie, Robert L. Palla and Christopher Gratton.

II. BACKGROUND

On December 23, 1998, Carolina Power & Light Company (Licensee or CP&L) filed an application for a license amendment, pursuant to 10 C.F.R. § 50.90, for the Shearon Harris Nuclear Power Plant (Harris or HNP) (Application). The Application sought approval to increase spent fuel storage capacity by adding rack modules to two spent fuel pools ("C" and "D") and placing the two pools into service. On January 13, 1999, the NRC published a notice of proposed no significant hazards consideration determination and opportunity for hearing. 64 Fed. Reg. 2237 (1999). On February 12, 1999, BCOC filed a request for hearing and petition to intervene.[1] The petition to intervene was granted by the Atomic Safety and Licensing Board (Board) by Memorandum and Order (Order), dated July 12, 1999. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-99-25, 50 NRC 25 (1999). The Board admitted two of BCOC's technical contentions for litigation. *Id.* at 38. On July 21, 1999, pursuant to 10 C.F.R. § 2.1109, the Licensee filed a request to invoke the hybrid hearing procedures of Subpart K and for oral argument.[2] On December 21, 1999, the NRC issued an Environmental Assessment (EA) and Finding of No Significant Impact (FONSI) for the license amendment application (Brief Exhibit A). 64 Fed. Reg. 71514 (1999).

Oral argument on the technical contentions was held on January 21, 2000 and on May 5, 2000, the Board issued a Memorandum and Order denying a hearing on the Technical Contentions. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-00-12, 51 NRC 247 (2000).

On January 31, 2000, BCOC filed a motion for admission of late-filed environmental contentions based upon the issuance of the December 21, 1999, EA and FONSI.[3] The motion was granted as to one contention, designated as Environmental Contention EC-6

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

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

[3] Orange County's Request for Admission of Late-Filed Environmental Contentions, January 31, 2000.

by the Board. *Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-00-19, 52 NRC 85 (2000).

In accordance with the Board's Orders and 10 C.F.R. § 2.1113, the Staff hereby submits its written summary of all the facts, data, and arguments known to the Staff and on which the Staff intends to rely at oral argument to refute the existence of a genuine and substantial dispute of fact as to Environmental Contention EC-6.

III. STATEMENT OF FACTS

Shearon Harris Nuclear Power Plant, Unit 1 (HNP) is a three-loop Westinghouse pressurized water reactor (PWR) operated by CP&L in Wake and Chatham Counties, North Carolina. The site was originally planned as a four unit site and the fuel handling building (FHB) was designed and constructed with four separate pools capable of storing spent fuel. At the present time, two pools, A and B, located at the south end, of the FHB are in use. The license amendment request seeks to place pools C and D, located at the north end of the FHB, in service by completing the SFP cooling system and installing high density spent fuel racks.[4]

The spent fuel storage system at the Shearon Harris plant is housed in the FHB, a reinforced concrete, seismically qualified structure located adjacent to the Unit 1 Containment Auxiliary Building, the Reactor Auxiliary Building, and the Waste Processing building. Affidavit of Gareth W. Parry, Stephen F. Lavie, Robert L. Palla and Christopher Gratton In Support of NRC Staff Brief And Summary of Relevant Facts, Data And Arguments Upon Which The Staff Proposes To Rely at Oral Argument on Environmental

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[4] *See* NRC Staff Brief and Summary of Relevant Facts, Data and Arguments Upon Which the Staff Proposes to Rely at Oral Argument on Technical Contentions 2 and 3, at 3-4, January 4, 2000.

Contention EC-6 (Affidavit) ¶ 43. The building is designed to protect its contents against natural phenomena, such as tornadoes, hurricanes, and floods. The FHB houses the four fuel pools, the north and south end spent fuel pool cooling water systems, the FHB ventilation system, and other systems, structures, and components relied upon to support refueling and fuel storage operations. Spent fuel from the operation of the Harris Unit 1 nuclear reactor is transferred to the FHB through the transfer tube located in the south end transfer canal and stored in spent fuel pool A or B. The Harris fuel storage system also accepts spent fuel from the Robinson and Brunswick nuclear stations. *Id.*

The fuel storage system consists of four seismically qualified, reinforced concrete fuel pools and a cask loading pit. The fuel pools and the cask pit are lined with stainless steel for compatibility with the pool water. Affidavit ¶ 44. Spent fuel is stored in seismically qualified storage racks at the bottom of the fuel storage pools. Transfer canals are provided between the cask pit and the pools so that spent fuel assemblies can be safely transferred underwater from one pool storage location to another. Isolation gates are provided between each pool and transfer canal. The gates are constructed of steel and have inflatable rubber seals to minimize leakage. The gates extend from the pool surface to approximately the elevation of the top of the fuel storage racks. *Id.*

Two spent fuel pool cooling and cleanup systems (SFPCCS) are provided to remove decay heat from the spent fuel stored in the four fuel pools. Affidavit ¶ 45. One SFPCCS services the south end pools (pools A and B), while the other system services the north end pools (pools C and D). The systems are designed to seismic Category 1 requirements, and the system pumps can be powered from on site emergency power. Each SFPCCS consists of two 100% capacity pumps, two heat exchangers, filters, and a purification loop with a demineralizer. While independent of each other, the cooling systems can share inventory through the main transfer canal. The Unit 1 Component Cooling Water System (CCWS) removes the decay heat from both the north end and south end fuel pool heat exchangers, and transfers the heat to the Service Water System. *Id.*

Each fuel pool cooling water system (north and south) is comprised of two 100% capacity cooling loops. Affidavit ¶ 46. The fuel pool cooling pumps are remotely operated from the control room. Control room and local alarms are provided to alert operators of abnormal water level and high temperature in the fuel pools. Should a loss of offsite power occur, the fuel pool cooling pumps can be restarted from the control room using emergency power provided by the emergency diesel generators. *Id.*

Each fuel pool cooling water system includes a non safety-related, non seismically qualified purification loop designed to remove impurities and lower the activity levels in the fuel pool coolant. Valving is provided between the cooling system and cleanup system to permit isolation of this non safety-related system. Affidavit ¶ 47.

Periodically, coolant makeup is required to offset the effects of evaporation, sampling, and fuel transfer activities. Several methods for adding additional coolant to the spent fuel storage system are available to operators. Affidavit ¶ 48.

The containment building at Harris is a steel-lined, concrete reinforced containment in the form of a vertical cylinder with a hemispherical dome and a flat basemat. Affidavit at ¶ 65. The basemat is a minimum 12 foot thick reinforced concrete slab. *Id.*

The Board's order admitting Environmental contention EC-6 for litigation in this Subpart K proceeding requires that the parties analyze the probability of the occurrence of a seven-step accident sequence involving the spent fuel pools. The sequence begins with a degraded reactor core accident caused by an unspecified initiator. As discussed below and in the Staff's Affidavit, in analyzing the sequence, the Staff has considered all identifiable initiators of a degraded core accident, utilizing both site specific and generic data and analyses. The second step is containment failure or bypass. Again, as discussed below and in the Staff's Affidavit, the Staff has considered numerous degraded core accidents that could lead to containment failure or bypass at the Harris site. The third step is loss of all spent fuel pool cooling and makeup. The Staff's affidavit addresses all identifiable degraded core accidents that could involve either concomitant or subsequent loss of cooling and/or makeup. In assessing steps four and five, extreme radiation doses precluding personnel access and inability to restart cooling or makeup due to extreme radiation doses, the Staff has considered all pathways of radiation that could preclude restarting cooling or makeup or that would lead to step six, loss of most or all pool water through evaporation. Step seven, initiation of an exothermic reaction in pools C and D has been conservatively given a probability of 1, for all events that result in uncovery of spent fuel, and has therefore not been analyzed.

The facts upon which the Staff relies with respect to EC-6 are set forth in the affidavit of Dr. Gareth W. Parry and Messrs. Stephen F. LaVie, Robert L. Palla and Christopher Gratton. The professional qualifications of the Staff's witnesses are summarized in their affidavit and set forth in detail in Exhibits 1 through 4 of the affidavit. Dr. Parry is qualified as an expert in Probabilistic Risk Assessment (PRA) by virtue of his education, experience, and demonstrated knowledge and skill regarding that subject. Mr. LaVie is qualified as an expert in radiation dose analysis by virtue of his training, experience, and demonstrated knowledge and skill regarding that subject. Mr. Palla is qualified as an expert in severe accident analysis and containment behavior by virtue of his education, experience, and demonstrated knowledge and skill regarding those subjects. Mr. Gratton is qualified as an expert in reactor system analysis by virtue of his education, experience, and demonstrated knowledge and skill regarding that subject.

IV. THE REGULATORY FRAMEWORK

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

This proceeding is governed by the hybrid hearing procedures of 10 C.F.R. § 2.1101 *et seq.* (Subpart K). Subpart K provides that its procedures may be used, at the request of any party, in contested proceedings concerning, *inter alia,* applications for a license amendment "to expand the spent fuel capacity at the site of a civilian nuclear power plant, through the use of high density fuel storage racks . . . ." 10 C.F.R. § 2.1103. [5] The procedures include submission of a detailed written presentation that must contain all the facts, data, and arguments known to the party and on which the party intends to rely at oral argument to support or refute the existence of a genuine and substantial dispute of fact. 10 C.F.R. § 2.1113(a). All supporting facts and data must be submitted in the form of sworn written testimony or other sworn written submission. *Id.* The written submissions are to be served simultaneously on all other parties. *Id.*

After considering the submissions and the oral arguments, the presiding officer will issue an order (1) designating any disputed issues of fact and law for hearing, and (2) disposing of any issues of fact or law not designated for hearing. 10 C.F.R. § 2.1115(a).

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[5] Subpart K was promulgated in order to implement Section 134 of the Nuclear Waste Policy Act of 1982 (NWPA). Pub. L. 97-425, January 7, 1983, 96 Stat. 2201, 42 U.S.C. § 10101. *See Carolina Power & Light Co.* (Shearon Harris Nuclear Power Plant), LBP-00-12, 51 NRC at 247, 254 (2000).

In designating issues for hearing, the presiding officer "shall identify the specific facts that are in genuine and substantial dispute, the reason why the decision of the Commission is likely to depend on the resolution of that dispute, and the reason why an adjudicatory hearing is likely to resolve the dispute." *Id.* As for the issues not designated for hearing, only a brief statement of the reasons for the disposition is required. *Id.*

Subpart K provides for a form of summary disposition procedure. 50 Fed. Reg. 41662, 41664 (1984). The burden of demonstrating the existence of a genuine and substantial issue of material fact is on the party requesting adjudication. *Id.* at 41667. As stated by this Board in its decision denying a hearing on the technical contentions, "with respect to a merits disposition of any substantive matter at issue in this proceeding (i.e., the admitted BCOC contentions), relative to the central subpart K issue of the existence of disputed material facts requiring an evidentiary hearing, 'the burden . . . [is] on the party requesting the adjudication.'" *Harris*, LBP-00-12, 51 NRC at 254-55.

In promulgating Subpart K, the Commission emphasized that the threshold for an evidentiary hearing is strict:

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

50 Fed. Reg. at 41667. Therefore, in this case, the burden of going forward and of demonstrating the existence of a genuine and substantial issue of material fact that can only be resolved by the introduction of evidence at an adjudicatory hearing is on the

Intervenor, BCOC. *See Harris*, LBP-00-12, 51 NRC at 255. In order for any issue to proceed to hearing, the Board must "identify the specific facts that are in genuine and substantial dispute, the reason why the decision of the Commission is likely to depend on the resolution of that dispute,[6] and the reason why an adjudicatory hearing is likely to resolve the dispute." 10 C.F.R. § 2.1115(a).

B. National Environmental Policy Act of 1969, 42 U.S.C. § 4321, et seq.

Contention EC-6 challenges the adequacy of the Staff's environmental review of the proposed license amendment under the National Environmental Policy Act of 1969, 42 U.S.C. § 4321, et seq. ("NEPA"). Congress enacted NEPA in order to compel Federal agencies to consider the environmental consequences of their actions prior to making agency decisions. NEPA requires Federal agencies to prepare an Environmental Impact Statement ("EIS") detailing the possible environmental impacts of any major Federal action that may significantly affect the environment. *San Luis Obispo Mothers for Peace v. NRC*, 751 F.2d 1287, 1298 (D.C.Cir. 1984), *cert. denied*, 479 U.S. 26 (1986). Preparation of an EIS ensures that the environmental goals set forth in NEPA are "infused into the ongoing programs and actions of Federal Government." *Marsh v. Oregon Natural Resources Council*, 490 U.S. 360, 371 n.14 (1989), *citing* 115 Cong. Rec. 40416 (1969) (remarks of Senator Jackson) ("Marsh"). *See also Louisiana Energy Services, L.P.* (Claiborne Enrichment Center), CLI-98-3, 47 NRC 77, 87 (1998) ("LES"). Specifically, for all proposed major Federal actions, section 102(2)(C) of NEPA requires agencies to include a detailed statement of the environmental impact of the proposed action, any adverse environmental

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[6] This criterion is far stricter than a finding that an issue is material pursuant to the summary disposition rules. *See* 10 C.F.R. § 2.749(d).

effects which cannot be avoided should the proposal be implemented, and the alternatives to the proposed action. 42 U.S.C. § 4332(2)(C).

Every major NRC licensing action which significantly affects the environment is a major Federal action subject to the requirements of NEPA. *Scientist Institute for Public Information v. AEC*, 481 F.2d 1079 (D.C.Cir. 1973). The NRC has implemented the NEPA requirements in its regulations at 10 C.F.R. Part 51. The regulations require the preparation of an EIS for certain actions meeting the criteria in 10 C.F.R. § 51.20. For those actions which do not meet these criteria, the NRC is required to prepare an environmental assessment ("EA"). 10 C.F.R. § 51.21. Under 10 C.F.R § 51.30, an EA must identify the proposed action and include a discussion of the need for the proposed action, alternatives to the action, and the environmental impacts of the proposed action. Once the EA is complete, the NRC must decide whether the impacts are significant and require the preparation of an EIS, or whether the impacts are insignificant. If the impacts are determined to be insignificant, the NRC will issue a Finding of No Significant Impact ("FONSI"). 10 C.F.R. § 51.31. The FONSI must identify the proposed action, state that the Commission has decided not prepare an EIS, and present the reasons why the proposed action will not have a significant impact on the environment. *Id.*

To aid in the interpretation of NEPA and its requirements, a rule of reason has been developed and applied in order to determine whether an agency has adequately analyzed the environmental consequences of a proposal for action. *Natural Resources Defense Council v. Morton*, 458 F.2d 827, 834 (D.C.Cir. 1972). The rule of reason, broadly stated, requires that:

> if the environmental aspects of proposed actions are easily identifiable, they should be related in such detail that the

> consequences of the action are apparent. If, however, the effects cannot be readily ascertained and if the alternatives are deemed remote and only speculative possibilities, detailed discussion of environmental effects is not contemplated under NEPA.

Environmental Defense Fund, Inc. v. Andrus, 619 F.2d 1368, 1375 (10th Cir. 1980). Agencies are not required to conduct a "crystal ball inquiry," but must comply with NEPA's requirements to the fullest extent possible. *Id.*

In cases where new environmental information is brought to the attention of an agency, application of the rule of reason must be based on the value of the new information. *Marsh,* 490 U.S. at 373. In determining whether an EIS is required, the agency should consider the significance of the new information and determine whether it presents "a seriously different picture of the environmental impact of the proposed project from what was previously envisioned." *Hydro Resources, Inc.* (2929 Coors Road, Suite 1010, Albuquerque, NM), CLI-99-22, 50 NRC 3, 14 (1999) (rejecting Intervenors requests for a supplemental EIS), *citing Sierra Club v. Froelke*, 816 F.2d 205, 210 (5th Cir. 1987). The new information must be considered within the context of the broad discretion granted to agencies by NEPA to keep their inquiries within appropriate and manageable boundaries. *LES*, 47 NRC at 103. This broad discretion ensures that NEPA will not be construed too broadly, which would result in available agency resources being "spread so thin" that agencies are unable to adequately pursue protection of the physical environment and natural resources. *Id.*

V. <u>THE ADMITTED CONTENTION</u>

In its August 7, 2000 Order, the Board admitted one environmental contention for litigation in this proceeding, designating it EC-6. *Harris*, LBP-00-19, 52 NRC 85. The contention reads:

> 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. . . . 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 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 accidents 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 addresses in an EIS.
>
> There are two respects in which the proposed license amendment would significantly increase the risk of an accident at Harris:
>
>> (1) CP&L proposes several substantial changes in the physical characteristics and mode of operation of the Harris plant. The effects of these changes on the accident risk posed by the Harris plant have not been accounted for in the Staff's EA. The changes would significantly increase, above present levels, the probability and consequences of potential accidents at the Harris plant.
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>> (2) During the period since the publication in 1979 of NUREG-0575, the NRC's Generic Environmental Impact Statement ("GEIS") on spent fuel storage, new information has become available regarding the risks of storing spent fuel in pools. This information shows that the proposed license amendment would significantly increase the probability and consequences of potential accidents at the Harris plant, above the levels indicated in the GEIS, the 1983 EIS for the Harris operating license , and the EA. The new

> information is not addressed in the EA or the 1983 EIS for the Harris operating license.
>
> Accordingly, the Staff must prepare an EIS that fully considers the environmental impacts of the proposed license amendment, including its 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.

Id. at 93-94.

In admitting the contention, the Board said: "the crux of the contention . . . is whether the accident proposed by BCOC in basis F.1 of the contention has a probability sufficient to provide the beyond-remote-and-speculative 'trigger' that is needed to compel preparation of an EIS relative to this proposed licensing action." *Id.* at 95. The Board admitted the contention but confined consideration of it to the seven-step accident sequence postulated by BCOC (discussed below) and the associated probability analysis. *Id.* at 98. The sequence, as admitted, reads:

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

Id. at 95.

The Board also requested the parties to address the following three questions in their written presentations:

> 1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on page 13 <u>supra</u>? The estimates

should utilize plant-specific data where available and should utilize the best available generic data where generic data is relied upon.

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

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

The Staff's analysis of the seven-step accident sequence, the probability analysis and the answers to the Board's questions are discussed below.

VI. ARGUMENT

A. Introduction and Summary of Argument

The license amendment application submitted by CP&L proposes to modify Harris to increase the spent fuel storage capacity by adding rack modules to spent fuel pools C and D. The Intervenor claims that the NRC's environmental analysis of the proposed license amendment fails to meet the requirements of NEPA because the Staff did not consider the accident sequence postulated by the Intervenor. Pursuant to NEPA and 10 C.F.R. § 51.21, the NRC Staff prepared an EA for the license amendment application. Brief

Exhibit B. The NRC prepared an EA because this license amendment action does not fall within the criteria for mandatory preparation of an EIS under 10 C.F.R. § 51.20.

The EA contained all the relevant environmental information required by NRC regulations. Brief Exhibit B. After identifying the proposed action requested by CP&L, the EA stated that the proposed action was necessary for the licensee to provide spent fuel storage capacity for the CP&L units at Brunswick, Robinson and Harris through the end of their current licenses. The EA then examined the relevant environmental impacts from the proposed license amendment, including radioactive waste treatment, gaseous radioactive wastes, solid radioactive wastes, radiological impacts, and accident considerations. The EA also discussed the alternatives to the license amendment, including shipment of the fuel to a permanent federal fuel storage/disposal facility, shipment of the fuel to a reprocessing facility, reduction of spent fuel generation, an alternative creation of additional storage capacity, and the no action alternative. *Id.* After consideration of the environmental impacts and the alternatives, the NRC issued a FONSI for the license amendment. *Id.*

Intervenor BCOC has asserted that the Staff's EA was inadequate, and that therefore the resulting FONSI was improper, because the Staff failed to consider the accident sequence postulated by the Intervenor's witness, Dr. Gordon Thompson. BCOC claims that this accident sequence would result in significant impacts to the environment and that the Staff should have prepared an EIS on the proposed license amendment. The Licensing Board has clarified the issues raised by this contention by posing the three questions stated above.

In its memorandum and order admitting Environmental Contention EC-6, the Board cited the Commission decision in *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee

Nuclear Power Station), CLI-90-4, 31 NRC 333 (1990). In that case, as here, the intervenors sought preparation of an EIS in conjunction with a request by the licensee to expand its spent fuel pool capacity. *Vermont Yankee* at 334. The sequence asserted was very similar to the one presented by EC-6 and had been rejected by the appeal board as remote and speculative for NEPA purposes. *Id.* In discussing the requirements of NEPA, the Commission said:

> What is important for purposes of NEPA consideration is the likelihood of occurrence of the accident in question. If the accident sought to be considered is sufficiently unlikely that it can be characterized fairly as remote and speculative, then consideration under NEPA is not required as a matter of law.

Id. at 334-35. The Commission remanded the matter to the Appeal Board for further proceedings to develop more information on "the plausibility or probability of the [accident sequence] at issue . . . " in order to determine whether the accident is remote and speculative. *Id.* at 335-36. The Commission further said that if the Appeal Board found that the probability was on the order of 1E-4 ($1x10^{-4}$), the case was to be returned to the Commission for further review, but otherwise, "the Appeal Board should modify or confirm its judgment as to the remote and speculative nature of the accident on the basis of the accident probability derived on remand." *Id.* at 335-36. In a later opinion, responding to the Appeal Board's request for clarification, the Commission stated that "future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual probabilities and accident scenarios being analyzed." *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), CLI-90-7, 32 NRC 129, 132 (1990).

As discussed more fully below, the Staff has analyzed the accident sequence postulated by BCOC in order to determine its best estimate of the overall probability of the accident sequence. The Staff's best estimate is that the probability of the accident sequence is extremely low, with an upper bound of $2x10^{-7}$/reactor year.[7] Affidavit ¶ 251. Although the Staff has provided this numerical estimate of probability, the Staff also notes that the probability may in fact be lower than this number, as many sources of conservatism were taken into account in developing the estimate. *Id.* at 124, 127-131, 206, 207, 245, 254-256. Based on its best estimate of the probability and its belief that the actual probability of the sequence may be lower, the Staff submits that, pursuant to NEPA requirements, the probability of the accident sequence is so low as to be remote and speculative. Since NEPA does not require consideration of remote and speculative impacts, the Staff's EA for the proposed license amendment is adequate and further environmental analysis is unnecessary.

There are no recent developments that suggest any modification in the estimation of the individual events in the seven-step sequence. In addition, nothing contained in the ACRS's letter of April 13, 2000 suggests that the probability of the sequence or its component events is greater than previously analyzed.

If the Licensing Board reaches the conclusion that the accident sequence is not remote and speculative and that further environmental analysis of the accident sequence is necessary, the Staff submits that the appropriate type of environmental analysis at this stage in the process would be a reconsideration of the original EA and FONSI for the

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[7] In this submittal, two forms of notation are used. So, for example, 2×10^{-7} may also be written 2E-07.

proposed license amendment. Should the preparation of an EIS be required, the proper scope should be limited to the environmental impacts from the proposed license amendment and the appropriate alternatives to the proposed action.

B. The Testimony/Affidavit/Declaration of BCOC's Witness, Dr. Gordon Thompson, Should be Afforded Little Weight

BCOC has proffered Gordon Thompson as its expert witness for Contention EC-6, The staff submits that BCOC has not adequately demonstrated Dr. Thompson's expertise in Probabilistic Risk Assessment or any other discipline related to the determination of the probability of the occurrence of the seven-step accident sequence that is the subject of EC-6.

It is well established that persons who seek to present expert testimony must be qualified to do so. In order to establish an expert witness's testimony as reliable, a party seeking to offer such a witness's expert opinion must show that the witness has the necessary qualifications to offer an expert opinion on the matter, or must be capable of questioning the soundness of the expert opinions of persons who do appear as witnesses in the proceeding. 10 C.F.R. § 2.733 (1999); *see e.g., Duke Power Co.* (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-669, 15 NRC 453, 475 (1982). This standard is at the heart of the Commission's requirement of reliability.

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

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

BCOC has provided little reason to believe that Dr. Thompson is qualified to serve as an expert witness herein. As borne out by his deposition, discussed below, Dr. Thompson admits that he does not possess the "'knowledge, skill, experience, training, or education' germane to" a resolution of the issue under consideration in this case - an analysis of the probability of the seven-step accident sequence. *McGuire*, ALAB-669, 15 NRC at 475.

During his deposition on October 16, 2000, attached hereto as Brief Exhibit A,[8] Dr. Thompson made the following statements regarding his proffered expertise:

> 1) He stated that he will be providing "an integrated picture of the steps involved in [the] accident sequence, drawing from literature, where available, and supported by calculations and judgement, where necessary. This task involves knowledge of reactor accidents, the release of radioactive material from reactor containments during such accidents, the effect of this radioactive material on the ability of operators to maintain the functioning of spent fuel pools and the equipment that supports the operation of the spent fuel pools, the behavior of a spent fuel pool when water is lost through evaporation, and the radioactive release that would ensue from the drying out of a pool. . . . .[T]his is a highly complex matter involving a great deal of analysis. There is no one person who could claim to be an expert on every detail of every step of that analysis, and *I do not claim to be such a person.* I do not believe any such person exists. And, therefore, my expertise, which I believe is sufficient for this proceeding is to provide, as a I said before, an integrated picture, drawing upon literature where it exists and then supported by calculation and judgement." (Dep. Tr. 7-8) (emphasis supplied).
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> 3) He claims that his expertise in reactor accidents is sufficient for the purpose of this proceeding. He has participated in a number of studies relating to reactor accidents and core damage (a source-term report (Dep. Tr.

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

12)). This affords him expertise in providing an integrated picture of the sequence. (Dep. Tr. 9). He has no training in analysis of reactor accidents. He has not worked with codes to analyze reactor accidents. He believes that his education in science and engineering, which did not include any courses regarding reactor accidents, "is sufficient to support the testimony he has to provide." (Dep. Tr. 10-11).

4) He has had no training or education relating to containment integrity, containment design, containment bypass. (Dep. Tr. 13-14). His contribution to the source-term report mentioned above was based on a survey of the literature and attendance at conferences and workshops. (Dep. Tr. 15-16).

5) He worked on an unused report on the probability and consequences of the release of contaminants related to a request by the owners of Seabrook to reduce the emergency planning zone. (Dep. Tr. 17-19). He did no original calculations, other than some for consequences. (Dep. Tr. 19). He has participated in a number of other studies, but has not performed any original calculations or accident analyses for any of those reports. (Dep. Tr. 20-26).

6) He is not qualified to provide a thorough review of CP&L's IPE, IPEE or PSA. (Dep. Tr. 33-34).

7) He disagrees with the Board's assessment of his expertise as to reactor technical issues as "policy oriented," (Dep. Tr. 45-49).

8) He claims that he does not need to perform a detailed analysis of the accident sequence, nor could he, but "in order to support the contention, we do not need to perform the analysis . . . . All that is necessary is to show that the use of a set of reasonable assumptions and supported by some scoping calculations shows that . . . the probability is characterized . . . in such a manner that . . . Preparation of an EIS is required." (Dep. Tr. 57-58).

9) He has no training as a health physicist. (Dep. Tr. 72-73).

10) He claims that he is qualified to "lead a team that would provide a thorough and comprehensive assessment of the onset of oxidation reaction in a draining pool. . . . [He]

> is qualified to specify the problem, to interact with the various specialists necessary, to identify the specialists necessary, the calculations and experiments that are necessary to review their findings, and to summarize these into a credible result." (Dep. Tr. 94).
>
> 11) He considers himself an expert in PRA for the purposes of this proceeding. (Dep. Tr. 115) He has had no training and little experience in that discipline.

Based on the sworn testimony of Dr. Thompson during his deposition, as set forth above, it is clear that he lacks sufficient expertise related to the analysis of the probability of his postulated accident sequence in Environmental Contention EC-6. He does not have sufficient expertise in the disciplines necessary to evaluating the seven-step sequence. The rationale for permitting expert testimony and permitting an expert to give conclusions and opinions based upon data and facts, and other experts' analyses, is to aid the trier of fact in reaching a decision. *See McGuire,* ALAB-669, 15 NRC at 475. Even though BCOC's witness, Dr. Thompson, acknowledges that analysis of his proposed scenario requires a team effort with input from experts in several disciplines, that he is not qualified to render an expert opinion on the entire sequence, and that, at most, he is qualified to provide an integrated picture of the work of experts in the necessary disciplines, BCOC is relying solely on Dr. Thompson. His testimony regarding the probability of the seven-step accident sequence should be given very little weight. Any opinions Dr. Thompson may render in this matter based upon his expertise as explicated in his deposition will be of little aid to the Board in rendering a decision on this environmental contention. *See McGuire*, ALAB-669, 15 NRC at 475, n. 48.

As demonstrated above, Dr. Thompson's qualifications as an expert witness for the issues relevant the analysis of the probability of the seven-step accident sequence are meager. Therefore, any conclusions he makes, opinions he renders, or other testimony related to this contention should be given very little weight by this Board.

C. The Probability of the Occurrence of the Seven Step Sequence is Low

The Staff utilized probabilistic risk assessment (PRA) methods to analyze the probability of the occurrence of the seven-step accident sequence proposed by BCOC. In order to analyze the sequence, the Staff assembled a team of experts, led by Dr. Gareth W. Parry, an acknowledged expert in PRA, and including members with expertise in severe accident analysis and containment behavior, radiological consequences, and reactor systems. The experts analyzed the sequence utilizing, among other things, site-specific data and PRA analyses submitted by CP&L, and the most recent generic information and analyses.[9]

PRA was utilized in performing this analysis because it is the best methodology to determine the probability of the sequence. The Commission has accepted the use of PRA methodology and, in fact, has encouraged the Staff and the industry to use PRA and "to expand the scope of PRA applications in all nuclear regulatory matters. . . ." "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," 60 Fed. Reg. 42622, 42628 (1995). The NRC has adopted a risk-informed

[9] As discussed above, Dr. Thompson admitted that he was not qualified to perform the analyses required to assess the probability of the sequence, as ordered by the Board. Dep. Tr. 7-8, 33-34, 57-58, 94. He would therefore, be performing scoping calculations. Dep. Tr. 10. The Staff submits that this failure to fully respond to the Board's order, if borne out in BCOC's submittal, will demonstrate (1) Dr. Thompson's limited expertise in this area and (2) BCOC's failure to meet the required burden to demonstrate the existence of a genuine and substantial issue of material fact as to its postulated seven-step sequence and the need for the preparation of an EIS. The Staff notes that Dr. Thompson, who proposed the seven-step sequence, apparently does not deem it necessary to fully answer the Board's questions concerning the sequence.

approach to regulation that uses PRA information in conjunction with other information to assess risk. In its Final Policy Statement regarding the use of PRA methods, the Commission said:

> PRA addresses a broad spectrum of initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for multiple and common cause failures. The treatment therefore goes beyond the single failure requirements in the deterministic approach. The probabilistic approach to regulation is, therefore, considered an extension and enhancement of traditional regulation by considering risk in a more coherent and complete manner.

60 Fed. Reg. at 42627. As this Board noted:

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

Harris, LBP-00-19, 52 NRC at 97. It is against this backdrop that the Staff has examined the probability of occurrence of the proposed seven-step accident sequence.[10]

PRA is used to analyze degraded core accidents, which occur when heat cannot be removed from the reactor core and it overheats. If cooling is not restored in time, the core will melt, potentially leading to failure of the reactor pressure vessel and possible containment failure or bypass, which, in turn, may lead to release of radionuclides into the

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[10] The Staff's qualitative analysis ends at step five (inability to restart any pool cooling or makeup systems due to extreme radiation doses) because the probability of the sequence going beyond that step approaches zero. But, due to the simultaneous filing requirements of the subpart K procedure, the Staff has addressed several other factors that are not necessary to its analysis, but which may be raised by the Intervenor's witness.

environment. Affidavit ¶ 9. PRA is an analytical approach used to analyze such accidents. *Id.*

A PRA analysis of degraded core accidents and containment failures was done for the Harris plant by CP&L. Affidavit ¶ 10. It is found in the Harris Individual Plant Examination (IPE), which has been updated in the Probabilistic Safety Study (PSA). *Id.* CP&L also analyzed external events in the Individual Plant Examination for External Events (IPEEE). *Id.* These documents address the first two events of the seven-step sequence. *Id.*

A level 1 PRA is used to analyze the causes and likelihood of degraded core accidents. Affidavit ¶ 12. The analysis utilizes logic models called event trees that identify the various scenarios that can occur after a challenge to normal operation and that result from combinations of successes and failures of the functions or systems that are required in response to those challenges. Event trees are supported by other logic models called fault trees that identify the combinations of equipment and personnel failures that lead to system or function failure. *Id.* The fault trees combined with the event trees are used to identify the combinations of equipment and personnel failures that result in each of the degraded core accidents. *Id.* Level 2 PRAs are used to analyze the consequences of degraded core accidents in terms of the impact on containment using a containment event tree. *Id.* ¶ 13.

PRA analysis involves the use of engineering judgment in constructing the analytic models, which are used to extrapolate from common situations to situations that have never occurred. Affidavit ¶ 14. Therefore there is variability in the results of PRAs performed by different analysts. *Id.* The Staff has recognized this variability as a source of uncertainty and requires that the sources of uncertainty in PRAs be identified and evaluated. *Id.* ¶15. A peer review is one method of increasing confidence in PRA results.[11] *Id.*

PRAs have been performed for all nuclear power plants in the United States. Affidavit ¶ 16. Although there are differences in the detailed results, there is a general consensus regarding the types of accidents that can cause degraded core conditions and the necessity for plant specific analyses. *Id.* The Staff requires that any assessment of risk be accompanied by both an assessment of the impact of the identified uncertainties and qualitative arguments that justify the case. *Id.* ¶ 19. Therefore, in this case, the issue of whether the seven-step accident scenario is "remote and speculative" will not be based on a number alone, but an understanding of the reasoning that went into that number. *Id.*

There are many postulated degraded core accidents, each with its own characteristics and frequency of occurrence. Affidavit ¶ 22. The conditional probability of containment failure or bypass given a degraded core accident is dependent on the characteristics of the accident sequence. *Id.* There are some degraded core accidents for which SFP cooling would be interrupted. *Id.* ¶ 23. If pool cooling is interrupted for long enough, the water in the pools would eventually heat up and evaporate or boil off.[12] *Id.* ¶ 24. If pool cooling or makeup is interrupted, it must be restored before the fuel is uncovered. If pool cooling or makeup are restored, then BCOC's sequence is terminated.

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[11] The Staff notes that this analysis and CP&L's analyses (IPE and IPEEE) have been subject to peer review. Affidavit ¶¶ 30, 34, 35.

[12] The probability of an exothermic reaction in the fuel pool (step seven) has been assumed, conservatively, as 1, given the loss of the pool water. Affidavit ¶ 29.

Id. The Staff concludes that for many of the scenarios analyzed for the seven step sequence, the pool cooling function is recoverable. *Id.* ¶ 25.

Prolonged interruption of cooling will require a means of makeup of coolant. *Id.* There are several methods of makeup available at Harris. *Id.* In assessing the availability of makeup methods, the staff analyzed steps four and five, which address extreme radiation doses preventing access and restoration of cooling and makeup. *Id.* ¶ 26.

The location and degree of radiological contamination is a function of the nature of the degraded core accident and the containment failure mode and location. *Id.* The dose to personnel in contaminated areas is a function of the time from release and the time spent in the contaminated area. *Id.* The Staff interprets the probability of step five occurring as the probability of failure to restart any pool cooling or makeup systems given the constraints imposed by radiological contamination. *Id.* ¶ 28. The Staff concludes, as a result of its analysis, that there are no accident sequences identified where ability to either restart pool cooling or initiate makeup after an accident was precluded by severe radioactive doses. The Staff concludes that the probability of the seven-step sequence is very low and is bounded by 2E-07/year. *Id.* ¶ 30.

In its analysis, the Staff first addresses the probability of a degraded core accident at Harris.

The probability of a degraded core accident is usually presented as a core damage frequency (CDF). Affidavit ¶ 32. The Staff assessed the CDF, including contributions from all initiating events in all phases of reactor operation with fuel in the reactor, including internal events (e.g. loss of coolant accidents - LOCAs) and internal floods, external events (e.g. earthquakes, fires - internal or external, high winds), and accidents at low power or shutdown. *Id.* ¶¶ 34-40. The Staff reviewed CP&L's IPE, IPEEE, PSA, and answers to Staff interrogatories. *Id.* ¶¶ 34-37. The Staff also reviewed NUREG/CR-5750, NUREG-1448, and SECY-00-0007. *Id.* ¶¶ 37-40. The Staff's analysis of the documents reviewed is discussed in the Staff's affidavit at ¶¶ 32- 41. The Staff estimates the core damage probability at the Harris plant, including contributions from both internal and external initiating events from full power and low-power and shutdown states, to be 1.2E-04 per reactor year. Affidavit ¶ 235. This figure was reached by taking the sum of the probabilities for internal events (7E-05), fires (1.1E-05), seismic events (1E-05) and shutdown (3E-05). *Id.* ¶ 41. While there is some uncertainty with this estimate, the staff believes that this is a reasonably conservative assessment. The contribution from internal initiating events in particular is likely to be conservative, since the frequency of initiating events has been shown, based on plant specific data, to be considerably lower than that assumed in the IPE. *Id.* ¶ 235.

The Staff next analyzed the sequences that may result in interruptions of SFP cooling, noting that not all sequences that interrupt cooling will lead to loss of makeup. Affidavit ¶¶ 42, 237. The conditional probability that the spent fuel pool cooling is interrupted will be dominated by common causes of both core damage and the interruption of spent fuel pool cooling. *Id.* ¶ 236. The joint probability of both events from independent causes will be very low because the probability of failure of a redundant, normally operating system, such as the spent fuel pool cooling system over a short time is very low. *Id.* Therefore, the Staff focused on dependent causes. The staff considered four categories of degraded core accidents that may lead to interruption of spent fuel cooling. *Id.* ¶ 49. The major difference between these groups is timing - when pool heatup starts . *Id.* ¶ 50.

In analyzing these groups of events, the Staff reviewed CP&L's answers to the Staff's interrogatories and the IPE. Affidavit ¶¶ 51-55. The results of the Staff's review and its conclusions are found at ¶¶ 51-57 of the Staff's affidavit. The Staff also considered the likelihood that should the containment fail, the release of steam and radionuclides into the plant auxiliary buildings might affect equipment necessary to maintain pool cooling, primarily the component cooling water system, and emergency power systems, and concludes that the likelihood of such a consequential loss of cooling is low. Affidavit ¶ 115, 239. The Staff concludes that the frequency of events that lead to a loss of pool cooling is estimated to be less than 1E-04 (1×10^{-4}) per reactor year, with a best estimate of 6.3E-05 per reactor year (approximately 6 times in 100,000 years). *Id.* ¶ 240. This figure is reached by taking the sum of the probabilities for internal events and flooding (5.25E-05), seismic events (1E-05) and fires (2.6E-07). *Id.* ¶ 54.

All the sequences that are initiated by a loss of offsite power also result in the interruption of several of the methods of makeup to the pools, since the demineralized water system is not powered by the emergency buses. Affidavit ¶ 241. However, methods that employ gravity feed and the use of the fire protection system with the diesel driven fire pump would still be available. Therefore, no scenarios have been identified that directly lead to loss of all cooling and makeup systems. *Id.*

For the majority of accidents, even if they lead to an initial disruption of SFP cooling, the function is recoverable. Affidavit ¶¶ 58, 242. The likelihood of recovering the cooling function before containment failure depends on the precise timing of events. *Id.* ¶ 243. Because there is a very large number of possible scenarios representing different time sequences of events, the Staff did not focus on assessing the probability of restoration.

However for the very late containment failures, there is a very high probability that makeup or cooling would be restored before containment failure. *Id.* ¶¶ 137, 243.

Recovery is dependent on accessibility of those locations needed to effect recovery and whether the necessary equipment has survived the accident. Affidavit ¶¶ 55, 244. The Staff concluded that, whatever the initiating event , in order to determine the long term viability of SFP cooling, it is necessary to determine whether the needed equipment will still be available after containment failure, and whether the necessary locations are accessible. *Id.* ¶ 57.

The Staff next turned to analysis of containment failure or bypass modes and release category characteristics.

Core damage progression and containment response is evaluated in the level 2 PRA, which addresses severe accident phenomena important to accident progression and containment behavior and provides insights into the mechanisms that could lead to containment failure or bypass. Affidavit ¶ 58.

CP&L performed a level 2 analysis in the IPE, in which it evaluated and quantified accident progression using a containment event tree and supporting deterministic calculations and sensitivity analyses. Affidavit ¶ 59. The Staff and its contractors evaluated the level 2 IPE and found it to be complete and the results reasonable. *Id.* ¶ 60.

In analyzing the containment failure and bypass modes, the Staff considered CP&L's IPE, the Staff review of the IPE, the PSA, the NUREG-1150 internal events analysis for the Surry and Zion plants and the current state of knowledge regarding severe accidents and containment performance. Affidavit ¶¶ 60-62. The Staff also considered the likelihood of various containment failure modes reported in IPEs for other similar plants. *Id.* ¶ 63.

The Staff's discussion of these documents and its analysis is contained in the Staff's affidavit at ¶¶ 63-100 and graphically presented in Table 1 of the Staff's affidavit.

Based upon its analysis, the Staff found that the containment would remain intact in 80% of the core damage sequences, therefore, finding a conditional containment failure probability of 20%. Affidavit ¶ 66. The Staff determined that the containment failure modes of most concern are the early and late containment failures. *Id.* ¶¶ 132-139. Their combined probability of failure is less than .1. *Id.*

The Staff analyzed the impact of containment failure on spent fuel cooling. First evaluated was the likelihood that releases from the postulated containment failure could cause concomitant failure of spent fuel cooling, focusing on two possible failures: the component cooling water (CCW) system and the emergency and normal switchgear, both located in the reactor auxiliary building (RAB). Affidavit ¶ 102. In conducting its analysis, the Staff reviewed containment failures addressed in CP&L's IPE and PSA. *Id.* ¶¶ 102-114.

The Staff concluded that a release from the containment will either not reach the CCW components and the switchgear or will reach them with insufficient energy to have an adverse impact on the equipment. Affidavit ¶ 115. The Staff concluded that the probability of a degraded core accident that leads to an interruption of the pool cooling function and a containment failure prior to restoration of pool cooling is bounded by 6.3E-06. *Id.* ¶¶ 116, 246.

The Staff next addressed the possibility of restart of SFP cooling or makeup systems. In doing so, they considered the time available to perform recovery actions, the timing of containment failure, the methods of makeup available and the doses expected in the areas required to be accessible. Affidavit ¶ 118-131.[13] In analyzing the time available, the Staff used calculations relating to pools A and B. Although A and B are not under consideration here, as they have already been licensed, due to the higher heat loads and, therefore, faster evaporation, the Staff conservatively used data from pools A and B to calculate the time available to take action to restore cooling or initiate makeup to the pools. *Id.* ¶¶ 127-131. The Staff concluded that for a majority of the accidents that result in cooling interruptions, the function is recoverable. *See e.g.* Affidavit ¶¶ 130-131, 133, 209-217. The Staff also analyzed the various methods available for makeup. *Id.* ¶¶ 146-154, Table 2.

The Staff has identified no scenarios that, in the time available to provide makeup, would prevent access to all the areas where operator action is needed to establish makeup, although the time for access might be restricted because of dose considerations. Affidavit ¶¶ 217, 247. For most scenarios, access to the plant to initiate several of the methods is possible. Thus, the Staff states that it can be argued that element (5) of the seven step scenario has a probability of essentially zero. *Id.* ¶¶ 195, 217. However, the Staff further analyzed the sequence given the time available, taking into account that, for most of the scenarios there are several easily implemented methods accessible for providing makeup, and even taking into account human reliability considerations. Based on this analysis, the Staff found that the probability of a more broadly defined sequence, namely one in which

[13] As noted in the Staff's affidavit, CP&L has plans to seek an amendment to raise the heat load in pools C and D. For purposes of this proceeding, the Staff calculated the effects of increased heat load on pools C and D and concluded that it would take approximately 254 hours or 10 days for the water to heat up and boil down to the top of the spent fuel storage racks, assuming no steps to restore cooling or supply makeup. Affidavit at ¶ 121.

the degraded core accident leading to a containment failure and a loss of spent fuel pool cooling for a long enough time that the water is evaporated so that the fuel is uncovered is very low. *Id.*

The Staff assessed the likelihood of the operating staff restoring cooling or initiating makeup to a pool following a severe core damage accident with containment failure or bypass utilizing Human Reliability Analysis (HRA). Affidavit ¶¶ 218-234. Based on the analysis, the Staff concluded that once the decision has been made on a method for makeup, the likelihood of success is high. *Id.* ¶ 233. The most likely cause of failure to restore pool cooling or start makeup systems is considered to be a failure in recognizing the need to take action. *Id.* However, in those accidents in which the containment is likely to have failed before the need to provide makeup is clear, the staff at the site would already be in the process of damage control, and no longer concerned with protecting the reactor, but of limiting radiological contamination, and the spent fuel pool would likely be an obvious target of concern. *Id.* While no HRA method has been constructed and calibrated to provide human error probabilities for such situations, for the purpose of this analysis, the Staff assigned a probability of .1 to the event that the restoration of pool cooling or provision of makeup is not successful, for those cases where the only access is to the operating floor of the FHB. *Id.* ¶ 234. The contribution of equipment unavailability or unreliability is negligible, but actions may be hampered by steam. *Id.* For those cases where there are several methods available, and no access or environmental problems, the likelihood of failure is much lower, but assumed for this purpose to be .01. *Id.*

Therefore, the Staff concludes, based on its analysis of the seven-step sequence, utilizing site specific data and generic analyses, that the total frequency of a sufficiently prolonged loss of cooling to pools A and B (the most limiting time period) resulting in the uncovering of the fuel is estimated to be 2E-07. Affidavit ¶ 251. This is the upper bound of probability for the seven-step sequence, since the sequence, as written, is more restrictive. *Id.* Finally, the Staff detailed the sources of uncertainty and has characterized and provided the basis for the conclusions. *Id.* ¶¶ 251-256. After much detailed analysis, the Staff determined that they could identify no scenarios that would prevent access to the areas required to restore cooling or initiate makeup. For most scenarios, several methods of establishing makeup would be available. Since methods of makeup would be available for all scenarios, the probability that the seven-step sequence could occur is essentially zero. If human reliability considerations are factored into the analysis, it is estimated that the upper bound of probability is 2E-07/reactor year. In reaching this conclusion, the Staff used information from CP&L's PRA models found in the IPE, IPEEE and PSA, to estimate the joint probability of the first three steps of the sequence. Affidavit ¶ 252. The three documents were peer reviewed and the IPE and IPEEE were reviewed by the NRC staff. They are also consistent with PRAs for similar plants. *Id.* Estimates used for seismic and shutdown contributions to core damage frequencies were based on information from other similar plants and are believed to be representative. *Id.* ¶ 253. The Staff focused on the major factors impacting the outcome of the sequence. *Id.* ¶ 254. The conclusion of the analysis is not very sensitive to the uncertainties for the reasons that follow. The Staff took a conservative approach to estimating the conditional probability of interruption of cooling that requires restart following containment failure. *Id.* ¶ 255. No credit was taken for recovery of cooling before containment failure for early and late containment failure modes. *Id.* No credit was taken for assuming that, for 60% of the accidents, the control room would remain habitable. *Id.* Finally, because the Staff has demonstrated that there are no scenarios that would prevent access to at least one method of pool makeup, the probability is essentially zero. Therefore, a precise estimate of the probability is not needed. *Id.* ¶ 256. However, successful termination of the sequence depends on operator action. Therefore, the final estimate takes into account the likelihood of failure of the operators to successfully implement makeup to the pools. *Id.*

The Staff's affidavit demonstrates that many conservatisms have been factored into the analysis. *See e.g.* Affidavit ¶¶ 66, 101, 124, 127-131, 139, 206, 207, 234, 235, 245. In addition, the Staff, in assessing the time available to take the actions required, used the most limiting times for heatup of the fuel pools (used data from pools A and B, did not take credit for heat sink, and assumed that the events occurred early in the fuel cycle). *See e.g.* Affidavit ¶ 226.

The level of detail of the Staff's analysis, as evidenced by the Staff's Affidavit and supporting documentation, demonstrates that the Staff has produced a detailed and thorough analysis of the seven-step sequence, utilizing site specific data and generic data and analyses. The Staff's analysis has been internally peer reviewed, makes conservative assumptions, identifies the uncertainties, and characterizes the conclusions. The analysis is based largely on peer reviewed, site specific and generic analyses. Finally, the analysis is based on the knowledge and judgment of a team of experts from a variety of disciplines. Therefore, the conclusions drawn from the analysis regarding the very low probability of the seven-step sequence are reliable, well supported by data and analyses and represent the experts' best estimate of the probability of the occurrence of the sequence.

C. The Seven Step Sequence is Remote and Speculative

Once the probability of the occurrence of the proposed accident sequence is established, it must be determined whether NEPA requires consideration of such a sequence. An analysis of relevant NEPA law demonstrates that the probability of the seven-step accident sequence in question is so low as to be remote and speculative. Since NEPA does not require consideration of remote and speculative impacts, the NRC's environmental analysis of the proposed license amendment is adequate and further analysis is unnecessary.

The burden of proof for establishing that this accident sequence will result in a significant impact to the environment, and is not remote and speculative, rests on the Intervenor. *See Citizen Advocates for Responsible Expansion v. Dole*, 770 F.2d 423 (5th Cir. 1985), *reh'g denied en banc,* 777 F 2d 701 (5th Cir. 1985); *State of Louisiana v. Lee*, 758 F.2d 1081 (5th Cir. 1985), *cert. denied*, 475 U.S. 1044 (1986). In addition, as discussed above, the burden of going forward and of demonstrating the existence of a genuine and substantial issue of material fact that can only be resolved by the introduction of evidence at an adjudicatory hearing is on the Intervenor, BCOC. *See Harris*, LBP-00-12, 51 NRC at 255.

NEPA does not require that an EIS be prepared for impacts that are considered by the agency to be "remote and speculative." *San Luis Obispo*, 751 F.2d at 1300; *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29, 32 (1989) (stating that agencies need not consider "remote and speculative risks" or "events whose probabilities they believe to be inconsequentially small"); *Yankee Atomic Electric Co.* (Yankee Nuclear Power Station), LBP-96-2, 43 NRC 61, 90 (1996) (stating "only accident scenarios that are not 'remote and speculative' need be the subject of a NEPA analysis"). *See also Public Service Electric and Gas Co.* (Salem Nuclear Generating Station, Unit 1), ALAB-605, 14 NRC 43 (1981); *Houston Lighting and Power Co.* (Allens Creek Nuclear Generating Station, Unit 1), ALAB-629, 13 NRC 75 (1981). Neither the courts, the Counsel on Environmental Quality ("CEQ") implementing regulations for NEPA, 40 C.F.R. § 1500 et seq., nor the NRC implementing regulations for NEPA , 10 C.F.R. § 51.1, et seq., have clearly defined the phrase "remote and speculative." The phrase has been used to describe a variety of different circumstances and accident types that are not subject to a NEPA analysis. The phrase "remote and speculative" has been used to draw parallels to events whose probabilities are inconsequentially small. *San Luis Obispo, 751 F.2d* at 1300; *Vermont Yankee*, ALAB-919, 30 NRC at 32. It has also been used synonymously with "insignificant matters, such as those without import, or remote effects, such as mere possibilities unlikely to occur as a result of the proposed activity." *Environmental Defense Fund, Inc. v. Corps of Engineers of the United States Army*, 348 F.Supp. 916, 933 (N.D.Miss. 1972).

Several significant categories of accidents to which the phrase "remote and speculative" has been applied are "Class 9," "severe," and "beyond design-basis" accidents. "Class 9" refers to a category of accidents used by the Commission, in the past, to designate the most severe accidents for purposes of NEPA analysis. *Long Island Lighting Co.* (Shoreham Nuclear Power Station), ALAB-156, 6 AEC 831, 834 (1973) (hereinafter "*Shoreham*"). "Accidents that contemplate 'sequences of postulated successive failure more severe than those postulated for the design basis of protective systems and engineered safety features' are variously termed 'beyond design-basis,' Class 9,' or severe accidents.'" *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-880, 26 NRC 449, 458 (1987) (citations omitted) (hereinafter "*Diablo Canyon*"). This class of accident represents an "indefinable number of conceivable types of accidents" that are more severe than "design basis accidents." *Shoreham* at 834-35. The occurrence of Class 9 accidents is theoretically possible, but the probability of their occurrence is so small that their environmental risk is extremely low. *Id.* It has been held, therefore, that NEPA does not require a discussion of Class 9 accidents unless there is a reasonable probability of occurrence. *Id.*; *San Luis Obispo*, 751 F.2d at 1301; *Diablo Canyon*, 26 NRC at 460-61, *Vermont Yankee Nuclear Power Corp.* (Vermont Yankee Nuclear Power Station), ALAB-869, 26 NRC 13, 37 (1987).

The NRC no longer uses Class 9 to describe accidents and to determine which accidents must be considered in the design of nuclear power plants. The term "beyond design-basis" has replaced Class 9 to describe severe accidents. *Florida Power and Light Co.* (St. Lucie Plant, Unit No. 1), LBP-88-10A, 27 NRC 452, 458 (1988). *See also* Final Environmental Statement related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2 (FES), NUREG-0972, p. 5-59 to 5-60. (Brief Exhibit C) . Beyond design-basis accidents are "by definition, highly improbable--i.e., remote and speculative--events." *Vermont Yankee*, 26 NRC at 30-31. For this reason, beyond design-basis accidents which do not have a reasonable probability of occurrence are not required to be considered in an EIS.[14]

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[14] In some cases, intervenors have asserted that the Commission's research and analysis of beyond design-basis accidents demonstrates that the Commission does not consider these accidents to be remote and speculative. This argument has been rejected by the NRC and the Federal courts. "The existence of ongoing research into beyond (continued...)

The NRC has also attempted to define which accidents are remote and speculative by attaching to the accident a number representing its probability. For example, in *Public Service Electric and Gas Co.* (Hope Creek Generating Station, Units 1 and 2), LBP-78-15, 7 NRC 642, 699 (1978), the Licensing Board held that accidents which are expected to occur with "probabilities less than 1×10^{-6}, based on a conservative calculation, may be disregarded in the design basis of a facility," and that the environmental impacts of such low probability accidents are so remote and speculative that consideration of them in a supplemental EIS is not required. *Id.* at 698-699. In another case, the Appeal Board, in denying a stay pending appeal in a spent fuel pool expansion case, implied that an estimate of probability of loss of spent fuel pool water between 3×10^{-5} and 1×10^{-10} made in a Brookhaven draft report rendered such an accident remote for NEPA purposes. *Pacific Gas & Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-877, 26 NRC 287, 293 (1987). As discussed elsewhere in this brief, in Vermont Yankee, the Appeal Board found that a probability of 1×10^{-4} for a postulated accident was remote and

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[14](...continued)
design-basis accidents . . . does not undercut the reasonableness of the Commission's view that such accidents nonetheless remain highly improbable and therefore beyond NEPA's mandate." *Pacific Gas & Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-880, 26 NRC 449, 459 (1987), *citing San Luis Obispo*, 751 F.2d at 1301. "To the extent that the Commission ever considers the environmental impact and risks of a beyond design-basis accident, it does so as an exercise of discretion under its 1980 NEPA Policy Statement." *Vermont Yankee*, ALAB-869, 26 NRC at 31, *citing San Luis Obispo*, 751 F.2d at 1301. If, as stated elsewhere in this brief, the probability of these accidents is so small as to be remote and speculative, then the NRC's decision to exclude them from its consideration of the environmental impacts of this proposed license amendment is reasonable. The Staff has never, in either an environmental analysis or in research, considered the accident sequence postulated by the Intervenors. This exclusion of the accident sequence from review is reasonable in light of the small probability of its occurrence.

speculative. *Vermont Yankee*, ALAB-919, 30 NRC 29. On appeal, the Commission remanded for further proceedings due to the failure of the Appeal Board to develop information regarding the likelihood of the accident sequence postulated by the intervenors. The Commission specifically vacated that portion of the Appeal Board's holding that "an accident with a probability on the order of 10^{-4} per reactor year is remote and speculative." *Vermont Yankee,* CLI-90-4, 31 NRC at 335. Yet the Commission did not completely reject the Appeal Board's determination, saying: "We are reluctant either to endorse or reject a holding that accidents of this probability should be considered remote and speculative . . . . ." *Id.* The Commission instructed the Appeal Board to return the matter to the Commission if it found an accident probability on the order of 1×10^{-4}, but otherwise to "modify or confirm its judgment as to the remote and speculative nature of the accident on the basis of the accident probability derived on remand." *Id.* at 336. Therefore, although the Commission rejected the Appeal Board's holding, it did not reject the idea that accidents with a probability of 1×10^{-4} could be remote and speculative for purposes of NEPA.[15]

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[15] Many of the numerical probability findings have been used in the safety arena, but are instructive for the environmental issues raised by BCOC's contention. For example, the Appeal Board has stated that an accident with a conservatively calculated probability of 10(-6) per year or a realistically calculated probability of 10(-7) per year did not have to be considered in designing a plant. *Florida Power and Light Co.* (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-603, 12 NRC 30, 45 (1980). In other words, accidents with a probability lower than 10(-6) per reactor year are "beyond design-basis" accidents, or "Class 9" accidents under NRC's previous accident categorization, because they do not have to be considered in the design of nuclear power plants. Although there are no NRC cases directly discussing similar numerical values for remote and speculative accidents under NEPA, and the considerations for NEPA are not necessarily co-extensive with the Atomic Energy Act, the Staff submits that the above numerical probabilities are appropriate when considering the scope of a NEPA analysis.

The FES for the Harris plant did consider a number of severe, beyond design-basis accidents. (Brief Exhibit C). The estimated probabilities of the accidents analyzed in the FES ranged from 3 x 10^{-6}/reactor year to 4 x 10^{-5}/reactor year. The FES did not include any accident scenarios which had a more remote probability of occurrence, indicating that an approximate threshold for inclusion into an EIS would be in the range of 1E-06 occurrences per reactor year.

The Staff's best numerical estimate of the probability of this accident sequence is an upper bound of 2 x 10^{-7}, which is considerably lower than for the accidents termed "remote and speculative" in the above cases and the accidents evaluated in the Harris FES. Based on this information and the qualitative factors discussed in the Staff's Affidavit, the Staff has concluded that the accident sequence postulated by the Intervenors is remote and speculative.

The seven-step accident sequence proposed for consideration by BCOC falls into the definition of "beyond design-basis" accidents. It is a sequence of postulated successive failures more severe than those postulated for the design basis of protective systems and engineered safety features. This type of accident does not need to be considered in an EIS unless there is a reasonable probability of occurrence. Similar accidents to the one described by the Board in this case have been found to be outside the scope of an EIS.

In 1988, in a case similar to this case, a licensing board refused to admit a contention submitted by the Intervenor based on the remote and speculative nature of the contention. The Intervenor alleged that expansion of the spent fuel pool at the St. Lucie facility would increase the probability of a radioactive release to the environment as a result of normal plant operation. *Florida Power and Light Co.* (St. Lucie Plant, Unit No. 1), LBP-88-10A, 27 NRC 452, 457 (1988). The Intervenor postulated the following severe accident scenario: 1) a cask drop; 2) structural failure of the spent fuel pool; 3) loss of coolant; 4) fuel rod zircaloy cladding fire; and 5) large radiation releases to the environment. *St. Lucie*, LBP-88-10A, 27 NRC at 458. The licensing board found that this accident was a "Class 9" or "beyond design-basis" accident, and that there was nothing to suggest that "the loss of pool coolant and zircaloy cladding fire scenario . . . is anything but a remote and speculative, beyond design-basis accident . . . NEPA does not require the consideration of such an event and an EIS need not be prepared." *Id.* at 459, *citing San Luis Obispo*, 751 F.2d at 1300-1301 (D.C.Cir. 1984). *See also Vermont Yankee,* ALAB-869, 26 NRC at 30 (It "would be anomalous to require for a license amendment an EIS addressing remote and highly speculative consequences, when there was no such requirement for the operating license itself." NEPA does not require consideration of beyond design-basis accidents "because they are, by definition, highly improbable — i.e., remote and speculative — events."); *Northeast Nuclear Energy Co.* (Millstone Nuclear Power Station, Unit No. 3), LBP-00-02, 51 NRC 25 (2000) (rejecting accident sequence similar to the seven-step sequence postulated herein).

As discussed previously. In addition to the low numerical probability of the accident sequence, a number of other reasons exist as to why the accident sequence postulated by the Intervenors is remote and speculative for the Harris plant. First, the Staff has concluded that not all degraded core accidents would lead directly to an interruption of spent fuel pool cooling at Harris. The Staff has not identified any scenarios that directly lead to the loss of all cooling and makeup systems. Additionally, the Staff has not identified any scenarios which would prevent access to all areas where operator action is necessary to provide makeup water. For most of the scenarios evaluated, the Staff determined that several methods of providing makeup water to the spent fuel pools would be available. Given that a number of methods will be available, even considering the effects of human reliability, the probability of the postulated accident sequence is low. The Staff has determined that, for the Harris plant, a high likelihood of success of one of these methods is expected. Finally, the Staff noted a number of sources of conservatism where credit was not taken in developing the probability of the accident sequence.

Based on the above case law, the Staff's determination of the probability of the occurrence of the seven-step accident sequence, and other factors that make this accident sequence very unlikely to occur, the sequence can be appropriately categorized as "remote and speculative." Since NEPA does not require agencies to consider remote and speculative accidents in their environmental analyses, the NRC's EA and FONSI for the proposed license amendment are adequate and further environmental analysis is unnecessary.

D. Conclusion as to EC-6

Based upon the analysis contained in the Staff's Affidavit, the Staff concludes that the probability of occurrence of the seven step sequence under consideration is very low, with an upper bound of 2×10^{-7}/reactor year, but is probably lower due to the conservatisms in the analysis, and concludes that there are no postulated scenarios that would preclude access to the fuel handling building for the licensee to restore cooling or initiate makeup by at least one method.

The Staff submits that, based upon the analysis and conclusions contained in the Staff Affidavit and discussed above, the seven step sequence is remote and speculative. Therefore, an EIS need not be prepared.

The Staff submits that there are no further facts that need to be developed or that require the introduction of evidence in an adjudicatory proceeding for resolution. There are no genuine and substantial disputes of material facts as to any aspect of Environmental Contention EC-6. Therefore, BCOC's request for hearing should be denied and the matter should be resolved in favor of the licensee.

VII. THE LICENSING BOARD'S QUESTIONS

In its Order admitting the late filed environmental contention, the Board asked the parties to respond to several questions. The Staff's responses to the specific questions are contained in this section.

1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on page 13 supra? The estimates should utilize plant-specific data where available and should utilize the best available generic data where generic data is relied upon.

As discussed above and in the Staff's Affidavit, the Staff's best estimate of the overall probability of the seven step sequence is that it is very low, with an upper bound of 2×10^{-7}/reactor year (2 occurrences in 10,000,000 years). This estimate is based upon the plant specific data and generic data discussed *supra* and in the Staff's Affidavit.

2. The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the ACRS's April 13, 2000 letter suggest that

> the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an exothermic reaction, greater than was assumed in the decade-old NUREG-1353)?

The most recent published study of beyond design basis accidents in spent fuel pools is the Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (TWG), February 2000 (http://www.nrc.gov/NRC/REACTOR/DECOMMISSIONING/SF/index.html). Neither this report nor NUREG-1353 has a direct relevance to the probabilities of the individual events in the seven-step accident sequence, since they do not address severe core damage accidents as an initiating event for the loss of spent fuel pool cooling. However, it should be noted that the TWG report is in substantial agreement with NUREG-1353 in recognizing the very rare, high ground acceleration earthquakes as being the major concern. Such major earthquakes are not an issue here.

The concerns expressed in the Advisory Committee on Reactor Safeguards (ACRS) April 13, 2000 letter do not impact the estimation of the probability of the seven-step accident sequence. In particular, the Staff has taken the position in this affidavit, that even if the probability of step seven, the chance of an occurrence of an exothermic reaction, given the first six steps of the sequence, is assumed to be 1, the probability of the sequence is low enough that its occurrence is considered remote and speculative. Furthermore, the issue raised by the ACRS in relation to the ignition temperature does not directly impact the probability of the sequence. In the TWG report, the ignition temperature is used to determine the age of fuel for which an exothermic reaction is no longer a concern. In reality, a refinement of the ignition temperature could impact the time to ignition once the fuel is uncovered. However, since the probability of an exothermic reaction is assumed to be 1, this has played no role in the Staff's analysis.

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

This question requests that the parties define the scope of the NRC's environmental analysis should the Board determine that the accident sequence postulated by the Intervenors is not remote and speculative. The Intervenors have asserted that the EA prepared by the NRC was inadequate because it failed to consider their accident sequence. According to the Intervenors, the proper remedy for this inadequacy is for the Licensing Board to order a full blown EIS of the Harris plant. The Staff submits that preparation of an EIS would be an inappropriate remedy at this stage of the process. If the Licensing Board determines that the accident sequence is not remote and speculative and that the NRC should consider the impacts of the sequence, the Board should require the Staff to reevaluate its EA and the resulting FONSI.

The FES for the Harris plant, issued in 1983, considered the environmental consequences of operation of the plant on water, land, air, endangered species, and a number of other areas. Brief Exhibit C. The FES did not address the specific accident scenario posed by the Intervenors, nor did it address the impacts caused by the use of spent fuel pools C and D. The FES did, however, consider both design-basis accidents and severe accidents. The severe accident sequences evaluated had probabilities ranging from 3.0×10^{-6} occurrences per reactor year to 4.0×10^{-5} occurrences per reactor year. Brief

Exhibit C, 5-61. A detailed evaluation of the potential environmental and radiological impacts from these accidents followed, including a discussion of the dose and health impacts of releases, economic and societal impacts, and releases to groundwater. Brief Exhibit C, 5-64 to 5-84. After consideration of all these impacts, the FES concluded that "[t]hese impacts could be severe, but the likelihood of their occurrence is judged to be small." (Brief Ex. C, 5-84).

The EA prepared for the proposed license amendment considered whether the use of spent fuel pools C and D would result in impacts greater than those considered in the FES for the Harris plant. (Brief Ex. B). The EA considered certain accident scenarios, and concluded that the additional spent fuel pools would not result in greater impacts than those previously considered. The EA did not consider the scenario at issue in this case. The appropriate remedy at this point in the case, should the Licensing Board determine that the accident sequence is not remote and speculative, would be for the Staff to consider the impacts of the scenario as new information, prepare a new EA and reconsider whether the FONSI remains valid. The Staff would consider the significance of the accident scenario, and determine whether it presents a "seriously different picture of the environmental impact" analyzed in the FES. *Hydro Resources, Inc.* (2929 Coors Road, Suite 1010, Albuquerque, NM), CLI-99-22, 50 NRC 3, 14, *citing Sierra Club v. Froelke*, 816 F. 2d 205, 210 (5th Cir. 1987). In order to make this determination, the Staff would assess the environmental impacts of the accident sequence. Once this assessment is complete, the impacts of this sequence would be compared to the impacts of the severe accident scenarios examined in the FES. If the impacts from this accident sequence are not greater than the impacts discussed in the FES, then the Staff may reissue the FONSI. If, upon further examination of the environmental risk of the accident sequence, the Staff concludes that the impacts are greater than those of the severe accidents analyzed in the FES, the Staff would then prepare an EIS for the license amendment. To the extent any impacts of the license amendment are the same as the impacts analyzed in the FES and there is no new information regarding those impacts, the EIS may reference the FES.

If the Staff determines that an EIS is necessary, or, alternatively, if the Licensing Board requires the Staff to perform an EIS, the scope of the EIS would be limited to any new impacts from the license amendment not previously evaluated. The scope of an EIS in a license amendment proceeding is not as broad as that conducted in prior NRC licensing proceedings. An EIS for the proposed license amendment should consider the extent to which the action under the proposed amendment will lead to environmental impacts not previously evaluated. In this regard, the Appeal Board has stated:

> Nothing in NEPA or in those judicial decisions to which our attention has been directed dictates that the same ground be wholly replowed in connection with a proposed amendment . . . . Rather, its seems manifest to us that all that need be undertaken in a consideration of whether the amendment itself would bring about significant environmental consequences beyond those previously assessed and, if so, whether those consequences (to the extent unavoidable) would be sufficient on balance to require a denial of the amendment application.

Northern States Power Co. (Prairie Island Nuclear Generating Plant, Units 1 and 2), ALAB-455, 7 NRC 41, 46 n. 4 (1978). *See also Florida Power & Light Co.* (Turkey Point Nuclear Generating Station, Units 3 and 4), LBP-81-14, 13 NRC 677, 684-85 (1981).

An EIS addressing the impacts from the accident scenario would be limited to the proposed license amendment and would not consider the impacts of operation of the Harris plant. *Burbank Anti-Noise Group v. Goldschmidt*, 623 F. 2d 115, 116 (9th Cir. 1980). *See* *also Consumers Power Co.* (Big Rock Point Nuclear Plant), ALAB-636, 13 NRC 312, 329 (1981); *Florida Power and Light Co.*, 13 NRC at 685.

As noted above, the FES for the Harris plant considered in detail the impacts from a number of severe accidents. Therefore, any new EIS would be limited in two ways. First, the EIS would be limited to the impacts from the opening of spent fuel pools C and D, and not to the operation of the Harris plant. Second, the EIS would be limited to consideration of the impacts caused by the accident scenario which are either greater than the impacts previously analyzed or which were not addressed at all in the FES.

In addition to the requirement that agencies consider the environmental impacts of the proposed action, NEPA also requires agencies to consider alternatives to the proposed action, but such consideration is limited by the rule of reason. *See Vermont Yankee Nuclear Power Corp. v. Natural Resources Defense Council, Inc.*, 435 U.S. 519, 551 (1978).

Under these circumstances, should an EIS for the proposed license amendment be required, it would likely include a discussion of the alternatives to the proposed licensing action, as would the EA.

Therefore, if the Board determines that further environmental analysis of the postulated accident sequence is necessary, the Staff would consider the impacts from the sequence in an EA. The EA would be limited to a determination of whether the impacts of the accident sequence are greater than, or different from, the impacts analyzed in the FES for the Harris plant. If the EA concludes that the accident sequence would not result in any new significant impacts, or in any significant impacts which have not already been analyzed in the FES, the Staff will reissue a FONSI for the proposed license amendment. If the EA concludes that the accident sequence would result in significant impacts not analyzed in the FES, the Staff would prepare an EIS. The EIS will be limited in scope to the new impacts from the proposed license amendment. An EIS for this license amendment should address only the impacts from the accident sequence and the alternatives to the proposed action.

VIII. CONCLUSION

Based upon the foregoing, the Staff submits that there are no genuine and substantial disputes of material fact as to any aspect of Environmental Contention EC-6, including the probability of the seven-step scenario or the need for the preparation of an EIS, and there is no issue raised by the contention that required the introduction of evidence in an adjudicatory proceeding for resolution.

Respectfully submitted,

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Dated at Rockville, Maryland
this 20th day of November 2000