CASE SCHEDULED FOR ORAL ARGUMENT SEPTEMBER 5, 2002

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

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

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

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

JOINT APPENDIX VOLUME III: pages 1168 through 1785

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Dated: June 4, 2002

JOINT APPENDIX

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<u>Record</u>

<u>Note regarding omissions from the Joint Appendix:</u> Due to their volume, counsel for CP&L has agreed to omit some materials from the Joint Appendix that were included in the record submittal of CP&L's evidentiary presentation of November 20, 2000. Please see Joint Appendix Vol. II for a listing. The Court is respectfully directed to the record to review the omitted materials.

November 20, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT REGARDING CONTENTION EC-6

VOLUME 2

EXHIBITS 2-5

001168

EXHIBIT 2

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

AFFIDAVIT OF ROBERT K. KUNITA

COUNTY OF WAKE	)	
	) ss:	
STATE OF NORTH CAROLINA	)	

I, Robert K. Kunita, being sworn, do on oath depose and say:

1. I am a resident of the State of North Carolina. I am employed by Carolina Power & Light Company ("CP&L") and work at the Harris Nuclear Plant ("HNP" or "Harris Plant" or "Harris") in the Spent Fuel Management Subunit of the Environmental and Radiation Control Unit. Presently, I am a Principal Engineer, Spent Fuel Management. My business address is 5413 Shearon Harris Road, New Hill, North Carolina 27562-0165.

2. I hold a Bachelor of Science degree in Physics from the Illinois Institute of Technology and a Masters of Science degree in Nuclear Science and Engineering from Carnegie Mellon University. Since graduation, I have been employed by the Bettis Atomic Power Laboratory ("Bettis") and CP&L. At Bettis, from 1966 to

1973, I was a member of the nuclear core design team for Admiral Rickover's Light Water Breeder Reactor Project, which subsequently ran successfully at the Shippingport Reactor. I performed computerized nuclear design calculations and participated in fuel design changes to optimize breeding while safely generating reactor power.

3. Since joining CP&L in 1973, I have held several positions of increasing engineering responsibility. From 1973 to 1975, I was in the Power Plant Engineering Section responsible for the nuclear fuel related systems for the planned South River Nuclear Power Plant. In 1975, I transferred to the Nuclear Fuel Section of the Fuel Department, where my responsibilities included interfacing with the Harris Nuclear Plant project on matters relating to nuclear fuel and I began participating in fuel examinations at the Robinson Nuclear Plant.

4. From 1977 to 1988, I was the Principal Engineer and head of the Surveillance and Accountability Unit. My responsibilities included assuring adequate nuclear fuel mechanical design, monitoring nuclear fuel mechanical performance, providing thermal-mechanical fuel analysis, planning for spent fuel storage and transportation, and providing fuel related support to CP&L's nuclear plants. During this time my focus on zircaloy clad fuel in-reactor performance, fuel examination efforts, and fabrication of zircaloy clad fuel continued. Fuel performance improved significantly over this period. Around 1983, I initiated the Robinson Dry Storage Demonstration project and from then until 1989, I was

involved in the resolution of technical concerns regarding the performance of zircaloy clad fuel in dry storage. I was also a member of the Technical Management Oversight Committee for the dry storage project and participated in a number of technical meetings related to spent fuel issues.

5. In 1989, I transferred to the Emergency Preparedness and Spent Fuel Management unit of the Operations and Environmental Support Department where I was responsible for planning and coordinating the implementation of CP&L's spent fuel management program, which included both dry storage and shipment of spent fuel. In approximately 1992, my function and I transferred back to the Nuclear Fuel Section and, in 1998, to the Harris Nuclear Plant.

6. Since 1977, I have represented CP&L on numerous Nuclear Energy Institute and Electric Power Research Institute committees dealing with various aspects of nuclear fuel. In my current position as the Principal Engineer, Spent Fuel Management, I continue to be responsible for matters relating to spent nuclear fuel. I am also a Professional Engineer registered in North Carolina. My resume is provided in Attachment A to this affidavit.

7. The purposes of this affidavit are to provide facts, data and my opinion on which CP&L relies in evaluating the postulated occurrence of a zirconium self-sustaining exothermic oxidation reaction in Harris spent fuel pools C and D following a postulated loss of most or all pool water through evaporation (*i.e.*, "Step 7" in the seven step sequence of events identified on page 13 of the

Licensing Board's Order dated August 7, 2000 ("Order")) and to address the Board's second question concerning NUREG-1353. First, I describe the principles of a postulated self-sustaining exothermic oxidation reaction of zirconium spent fuel cladding. Second, I discuss the literature survey I conducted to research the likelihood of a self-sustaining exothermic oxidation reaction of zirconium spent fuel cladding occurring in the Harris spent fuel pools. Third, I describe the application of the information obtained in my literature survey to the spent fuel to be stored in Harris spent fuel pools C and D and the analyses I performed to establish that a self-sustaining exothermic oxidation reaction of zirconium spent fuel cladding is very unlikely in the Harris pools. Finally, I provide my conclusions on the unlikely occurrence of "Step 7" in the seven step sequence of events identified in the Board's Order.

PRINCIPLES OF THE EXOTHERMIC OXIDATION REACTION OF ZIRCONIUM SPENT FUEL CLADDING

8. Zircaloy, like most metals, undergoes an oxidation reaction in an air environment. This oxidation reaction is exothermic, meaning that the reaction releases heat. The oxidation rate, and, therefore, the rate at which heat is released, increases as the temperature of the zircaloy increases. At temperatures less than several hundred degrees Celsius, the exothermic oxidation reaction occurs very slowly.

9. The temperature of spent fuel zircaloy cladding is determined by the balance between the rate at which heat is generated in the fuel and the rate at which heat is transferred from the fuel. If the heat generation rate is greater than the rate at

which heat is transferred from the fuel, the temperature rises. If the generation rate is the same as the heat transfer rate, the temperature is constant, and the temperature decreases if the heat generation rate is less than the heat transfer rate.

10. The primary contributor to the heat generation rate in spent fuel is radioactive decay of material in the fuel, referred to as decay heat. The heat input from the spent fuel, also known as spent fuel decay heat, is primarily a function of the combination of the reactor power level, the burnup of the spent fuel, in megawatt-days per metric ton of fuel (MwD/Mtu), and the age (or "decay time") of the fuel. The decay heat rate drops drastically with time after the fuel is discharged from the reactor. Approximately five years after discharge from the reactor, the decay heat rate of the old, cold spent fuel is a small fraction of the decay heat rate of the same fuel when it was first stored in the spent fuel pools.

11. It is possible that in some conditions of very high cladding temperatures, the oxidation rate and the corresponding heat generation from the exothermic reaction can become a significant heat source, which, when added to the decay heat from the fuel, can contribute to a further increase in temperature. If the increase in heat generation rate due to the exothermic oxidation reaction exceeds the heat transfer from the fuel, temperatures continue to increase, causing further increase in oxidation reaction rate. This condition is referred to as self-sustaining exothermic oxidation, and is the focus of step 7. The clad temperature at which the self-sustaining oxidation reaction occurs is referred to as the critical cladding

oxidation temperature. The result of a self-sustaining oxidation reaction is likely to be loss of clad integrity and loss of fuel pellet containment in the fuel assemblies.

12. Spent fuel cladding below the critical cladding oxidation temperature continues to oxidize, but a protective oxide layer forms on the cladding and slows down the oxidation rate. The oxidation reaction proceeds at such a slow rate that more than sufficient time is available to reestablish cooling of the spent fuel before cladding damage resulting in exposure of spent fuel could occur.

13. The mechanisms for heat transfer from spent fuel are conduction, convection, and radiative heat transfer. Conduction and convection dominate heat transfer until clad temperatures reach several hundred degrees Celsius. Some early analyses assumed spent fuel pools were completely dry when calculating convection heat transfer, but later studies included sufficient detail to model the temperatures with water above the bottom of the fuel racks, which obstruct the free flow of air into the bottom of the fuel assemblies.

RESULTS OF A SURVEY OF THE AVAILABLE LITERATURE

14. I performed a survey of the publicly available literature regarding the potential for the initiation and propagation of a self-sustaining exothermic oxidation reaction involving spent fuel cladding following the partial, or complete, loss of water from a spent fuel pool.

15. In conducting my literature survey, I searched a number of sources to identify

documents addressing a self-sustaining exothermic oxidation reaction in spent nuclear fuel. My literature survey identified as relevant a total of seventeen (17) documents, which are listed in Attachment B to this affidavit.

16. My literature survey did not identify any analysis that reported a zirconium cladding oxidation temperature any lower than 800°C. Numerous studies report that the critical cladding oxidation temperature of zirconium spent fuel cladding is about 900°C. While the Advisory Committee on Reactor Safeguards ("ACRS") has indicated that the presence of zirconium hydrides on spent fuel may lower the critical cladding oxidation temperature, I did not identify any analysis that indicated zirconium hydrides would lower the critical cladding oxidation temperature below 800°C. NUREG/CR-5597 shows the onset of rapid zircaloy oxidation at 1500°K (1227°C). A table of reported critical cladding oxidation temperatures and the associated reports is presented in Attachment C to this affidavit.

17. Actual spent fuel has been heated up in air to a temperature of approximately 800°C under controlled laboratory conditions (see, Attachment B, reference 7). No zirconium self-sustaining exothermic oxidation occurred even when the spent fuel was heated to approximately 800°C in an air environment. This experimental result is consistent with the analyses reporting 800°C as a conservative lower bound for the critical cladding oxidation temperature of zircaloy fuel cladding.

APPLICATION OF AVAILABLE INFORMATION TO THE HARRIS SPENT FUEL POOLS C AND D

18. The residual or decay heat in nuclear fuel assemblies decreases rapidly after the fuel assembly is discharged from the reactor. Such decay heat is often described on an assembly basis in terms of the number of kilowatts of thermal energy given off (*i.e.*, in units of kilowatts per assembly). It can also be expressed on a per metric ton basis. This is determined by dividing the assembly kilowatts by the metric tons of uranium in the assembly. This kilowatts per metric ton is also referred to as a "specific heat" and is often used to make comparisons between different fuel types which differ in physical size and amounts of uranium.

19. Zircaloy self-sustaining exothermic oxidation does not occur below a fuel clad temperature of 800°C. This temperature will not be reached unless the specific heat is above some minimum value. For high burnup fuel stored in high density racks, the NRC has conservatively selected, in its "Technical Study on Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants" (Attachment B, reference 5), three (3) kilowatts per metric ton as this minimum value. This value was previously six (6) kilowatts per metric ton for lower burnup fuel stored in more widely spaced storage racks.

20. Harris spent fuel pools C and D will only store spent fuel aged five years or more out of the reactor. It is anticipated that all spent fuel shipped in the future from Brunswick and Robinson to Harris will be stored in spent fuel pools C and D.

21. Spent fuel from the Brunswick and Robinson plants that is to be stored in Harris

spent fuel pools C and D is transported to Harris in CP&L's transportation cask. The existing transportation cask has strict heat load limits on the spent fuel that can be transported in the cask. The transportation cask Certificate of Compliance 9001 limits the maximum heat load of spent fuel shipped in the cask to a peak of 5,725 Btu/hr/PWR and 2,225 Btu/Hr/BWR which is equivalent to 3.7 Kw/Mtu for PWR fuel and 3.6 Kw/Mtu for BWR fuel. As a practical matter, these limits have not been approached and the peak assembly heat load of spent fuel transported to Harris spent fuel pools A and B was 2.5 Kw/Mtu for PWR fuel and 2.9 Kw/Mtu for BWR fuel. The peak assembly specific heats, at the time of shipment, for the shipments made to Harris pools A and B are shown in Figures 1 and 2 for PWR and BWR fuel respectively (Attachments D and E).

22. Figure 3 shows the decreasing specific heat curves for the PWR fuel stored in Harris spent fuel pools A and B (Attachment F). Some of this spent fuel will be moved to pools C or D in the future to make room for spent fuel discharges from the Harris reactor. Present projections show that the Robinson spent fuel will be 15 to 20 years old at the time it is moved to pools C or D. Due to its age, this spent fuel does not have a high enough specific heat to cause a self-sustaining exothermic oxidation reaction in pools C or D.

23. Figure 4 shows the decreasing specific heat curves for the BWR spent fuel stored in Harris spent fuel pools A and B (Attachment G). There are no present plans to move this spent fuel to pools C or D. This fuel has already cooled between 6 and

23 years. Due to its age, this spent fuel will not have a high enough specific heat to cause a self-sustaining exothermic oxidation reaction in pool C or D.

24. Figure 4 also applies to BWR fuel assemblies to be shipped from the Brunswick plant and then unloaded at Harris and stored in pools C or D. Since such fuel will have at least 5 years of cooling prior to shipment, this fuel will also not have a high enough specific heat to cause a self-sustaining exothermic oxidation reaction in pool C or D.

25. Figure 5 shows the decreasing specific heat curves for the high burnup Harris PWR fuel (Attachment H). Some Harris fuel with burnups over 40,000 MwD/Mtu was added to Harris spent fuel pools A and B beginning in 1991. Such spent fuel is not presently projected to be moved to pools C or D until 2011 or later; hence, such spent fuel will have cooled about 20 years. Due to its age, this spent fuel will not have a high enough specific heat to cause a self-sustaining exothermic oxidation reaction in pools C or D.

26. Anecdotal evidence exists that shows that zircaloy self-sustaining exothermic oxidation does not occur for cooled spent nuclear fuel. Between late 1977 and early 1981, CP&L shipped 290 PWR fuel assemblies from Robinson to Brunswick in over 40 shipments using air coolant in the shipping cask. At the time of shipment, this fuel had cooled between 2.7 and 6.5 years. There is no evidence that there was anything unusual about these assemblies when they were unloaded after receipt at Brunswick.

27. As shown in Figure 6, the amount of Ruthenium-106 decreases rapidly after the spent fuel assembly is removed from the reactor (Attachment I). This is because Ruthenium-106 has a half life of approximately one year; hence, half of the Ruthenium decays away every year. Only fuel aged for five years or more will be placed in pools C and D. By the time that pool C is filled, the average age of the fuel will be 23 years for PWR fuel and 10 years for BWR fuel. By the end of the Harris operating license (*i.e.*, 2026), the average age of the fuel in pool D will be 23 years for PWR fuel. No BWR fuel is planned to be stored in pool D. In 2026, the average age of the fuel in pool C will be 30 years for the PWR fuel and 22 years for BWR fuel. Thus, any evaluation of the amount of Ruthenium in pools C and D is not particularly meaningful.

28. Even if a small number of spent fuel assemblies in Harris spent fuel pools C and D could potentially sustain a zirconium self-sustaining exothermic oxidation reaction, it would be unlikely to propagate to adjacent assemblies because they have a heat load far too low to sustain a the reaction. Thus, even if a zirconium self-sustaining exothermic oxidation reaction could occur in Harris spent fuel pools C and D, its extent would likely be extremely limited because of the large quantity of extremely old, cold spent fuel stored in the pools and the small, if any, amount of undecayed Ruthenium remaining. Thus, it would appear that the consequences of a postulated zirconium exothermic oxidation reaction in spent fuel pools C and D would be bound by the consequences on the severe reactor

accident that initiated the scenario set forth in the Board's Order.

IMPACT OF RECENT DEVELOPMENTS ON NUREG-1353 ESTIMATES

29. I have reviewed the documents listed in Attachment J to this Affidavit to evaluate their impact on two estimates contained in NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'" (1989).

30. To the extent that any NUREG-1353 estimated value is applicable to the specific seven step scenario contained in the Board's Order, it is my opinion that the data or models that have been reported since the publication of NUREG-1353 do not suggest any substantive modification of those values. It is also my opinion, however, that, with the exception of the values associated with the probability of a self-sustaining exothermic oxidation reaction occurring in spent fuel following loss of spent fuel pool water level, the estimated values in NUREG-1353 do not appear applicable to the scenario postulated in the Order.

31. Regarding the probability of a self-sustaining exothermic oxidation reaction being greater than assumed in NUREG-1353, it is my opinion that this is clearly not the case for PWR spent fuel, as NUREG-1353 assumes a probability of 1.0 for this event. As to BWR spent fuel, I did not find in my literature search any basis for changing the probability value of 0.25.

CONCLUSIONS

32. As long as the heat output of spent fuel is less than the available heat removal

capability, the spent fuel will remain cool, and no self-sustaining zirconium exothermic oxidation reaction will occur.

33. For spent fuel with heat outputs less than the limits identified in the literature, no self-sustaining zirconium exothermic oxidation reaction will occur even if spent fuel pool water inventory is lost because the available energy is insufficient to initiate and sustain the reaction.

34. For spent fuel with a heat output above the identified limits, it is unclear whether a self-sustaining exothermic oxidation reaction will occur.

35. I conclude, therefore, that because of the low heat load in the old, cold spent fuel to be stored in Harris spent fuel pools C and D, it is highly unlikely that the spent fuel in pools C and D could sustain a zirconium self-sustaining exothermic oxidation reaction, even if most or all of the water in pools C and D is lost through evaporation.

36. I also conclude that in the highly unlikely event that a zirconium self-sustaining exothermic oxidation reaction were to occur in Harris spent fuel pools C and D, its extent would be extremely limited because most of the spent fuel to be stored in pools C and D will be far too old and cold to propagate the reaction. The consequences would certainly be bound by the consequences of the postulated degraded core accident with containment bypass that is the postulated initiator of the seven step sequence in the Board's Order.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2000.

Robert K. Kunita

Robert K. Kunita

Subscribed and sworn to before me
this 15 day of November, 2000.

Darlene S. Yarborough

My Commission expires: 2-21-2005

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

001182

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New Hill. NC 27562-9217
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Robert K. Kunita

Professional Experience

1973 - Present Carolina Power & Light Company

Principal Engineer – Spent Fuel Management

During my 27 years with Carolina Power & Light, I have worked in the Power Plant Engineering Section, the Nuclear Fuel Section, and the Emergency Preparedness & Spent Fuel Management Sections, all of which were in the Corporate Offices in Raleigh, NC. I have worked for the past two years at the Harris Nuclear Plant located in New Hill, NC in the Spent Fuel Management Subunit of the Environmental and Radiation Control Unit.

My experience covers a broad range of nuclear fuel related items from reactor systems interfaces, fuel design, fuel fabrication, nuclear material accountability, and spent fuel management. I was responsible for and accomplished reviews of system designs and NRC license application submittals, development and implementation of nuclear fuel fabrication surveillance plans, establishment and maintenance of a nuclear material accountability program, development of a dry spent fuel storage demonstration project which was successfully implemented, preparation of implementation of spent fuel shipping emergency exercises, and development of a corporate spent fuel management plan.

I have reviewed documents from the NRC, NEI, EPRI, etc. for technical adequacy and impact on CP&L and I have represented CP&L on numerous NEI and EPRI spent fuel committees.

1966 - 1973 Bettis Atomic Power Laboratory West Mifflin, PA

Associate Engineer through Senior Engineer

I worked for 7 years at the Bettis Atomic Power Laboratories, which was run by Westinghouse for the Naval Reactors Program. I was a member of the nuclear core design team for Admiral Rickover's Light Water Breeder Reactor Project, which subsequently ran successfully at the Shippingport Reactor. I performed computerized nuclear design calculations and participated in fuel design changes to optimize breeding while safely generating reactor power.

Education

Carnegie Mellon University Pittsburgh, PA

- 1973 M. S. Nuclear Science and Engineering

Illinois Institute of Technology Chicago, IL

- 1966 B.S. Physics

Registration	Registered Professional Engineer • North Carolina, PE #007015
Awards	1993 CP&L Quality Achievement Award
Professional Memberships	American Nuclear Society Eastern Carolinas Section of the American Nuclear Society, past membership chairman and treasurer.

Attachment B

References

1. N. A. Pisano, F. Best, A. S. Benjamin and K. T. Stalker, "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool", Draft Report, January, 1984.

2. A. S. Benjamin, D. J. McCloskey, D. A. Powers, and S. A. Dupree, "Spent Fuel Heatup Following Loss of Water During Storage", NUREG/CR-0649, March, 1979.

3. V. L. Sailor, K. R. Perkins, J. R. Weeks, and H. R. Connell, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82", NUREG/CR-4982, July, 1987.

4. R. J. Travis, R. E. Davis, E. J. Grove, M. A. Azarm, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants", NUREG/CR-6451, August, 1997.

5. D. Jackson, T. Eaton, G. Hubbard, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants", Draft, February, 2000.

6. A. B. Johnson, Jr., J. C. Dobbins, F. R. Zaloudek, E. R. Gilbert, and I. S. Levy, "Assessment of the Integrity of Spent Fuel Assemblies Used in Dry Storage Demonstrations at the Nevada Test Site", PNL-6207, July, 1987.

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

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11. C. F. Boyd, "Predictions of Spent Fuel Heatup After a Complete Loss of Spent Fuel Pool Coolant", NUREG-1726, July, 2000.

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14. A. W. Cronenberg, "In-Vessel Zircaloy Oxidation/Hydrogen Generation Behavior During Severe Accidents," NUREG/CR-5597, September, 1990.

15. E.T. Burns, D. E. True, V. M. Andersen, "A Review of Draft NRC Staff Report: DRAFT Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants", August, 1999

16. D.A. Powers, "Draft Final Technical Study of Spent Fuel Pool Accidents Risk at Decommissioning Nuclear Power Plants", letter to R. A. Meserve (NRC), April 13, 2000.

17. D. A. Powers, "Proposed Resolution of Generic Safety Issue-173A, 'Spent Fuel Storage Pool For Operating Facilities'", letter to R. A. Meserve (NRC), June 20, 2000.

Attachment C

Table of Reported Zircaloy Temperatures

Temperature	Reference (Attach B)	Description	Comment
1500^0K (1227^0C)	14, page 6-9	"Start of rapid zircaloy oxidation by H_2O uncontrolled temperature escalation"	
Around 900^0 C.	2, page 47	"The temperature at which clad oxidation becomes self-sustaining is a function of the storage configuration, but tends to occur around 900^0 C."	
About 900^0 C.	3, page 50.	"If the decay heat level is sufficient to heat the rods to about 900^0 C (1650^0 F) the oxidation becomes self-sustaining. That is, the exothermic oxidation reaction provides sufficient energy to match the decay heat contribution and the temperature rises rapidly."	
565^0 C	4, page 3-4	'The Workshop on Transportation Accident Scenarios estimated incipient clad failure at 565^0 C with expected failure at 671^0 C. presumably based on expert opinion. Given that the large seismic event is the dominant contributor to the configuration 1 initiator, it is likely that it would take a prolonged period of time to retrieve the fuel, repair the spent fuel pool or establish an alternate means of long-term spent fuel storage. Therefore, we presume there will be a significant period of time that the fuel will be exposed to air. On this basis, BNL has chosen a temperature of 565^0 C as the critical cladding temperature."	Incipient clad failure is not the onset of self-sustaining zirc reaction; it is the onset of clad swelling.
800^0 C	5, page A1-1	"The onset of rapid oxidation may occur as low as 800^0 C.	References 3.
About 900^0 C	9, page ES-1	If the decay heat level is high enough to heat the fuel rod cladding to about 900^0 C (1650^0 F) the oxidation becomes self-sustaining, resulting in Zircaloy cladding fire. Propagation of the Zircaloy cladding fire to	

		older adjacent assemblies is likely if the decay heat level in an older adjacent assembly is high enough to heat that assembly to within 100 to 200^0 C (200 to 400^0 F) of the self-sustaining oxidation temperature. Although propagation of a Zircaloy cladding fire to one or two year old fuel by only thermal radiation can occur, the older fuel would have to be next to the hottest assembly."	
800^0 C	10, page 4	"The working group is reviewing the temperature criteria used in the spent fuel analysis and the preliminary results indicate that a maximum allowable temperature of 800^0 C may be acceptable if certain analysis conditions are met. The conditions for applying this criteria would include demonstrating that the maximum calculated temperature, including uncertainties, remained below the temperature limit, that higher temperature effects are accounted for, and that a release of the radionuclides in the gap between the clad and the fuel is not a concern. The 800^0 C temperature limit is based on the lowest temperature for the onset of self-sustaining zirconium oxidation identified by the GSI 82 studies."...	
1600^0 F (871^0 C)	13, page 354	"It was concluded that a new evaluation model for Zircaloy oxidation should be applicable above 1144^0 K (1600^0 F), which is the temperature at which the Zircaloy oxidation rate becomes appreciable." "Measurements of the oxidation kinetics from prefilmed cladding and from specimens subjected to anisothermal conditions permit one to obtain some measure of the margin associated with different representations of the Zircaloy oxidation kinetics. The evidence shows that the presence of a prefilm significantly inhibits subsequent oxidation." [Prefilm is preoxidation or anomalous oxidation, see page 350].	This report references instanc where zircaloy w oxidized at 927 ^{0} C, 1316^0 C, 1038^0 C, 1000 to 1690^0 C, 1300 to 1750^0 C. There is mention of exothermic oxidation.

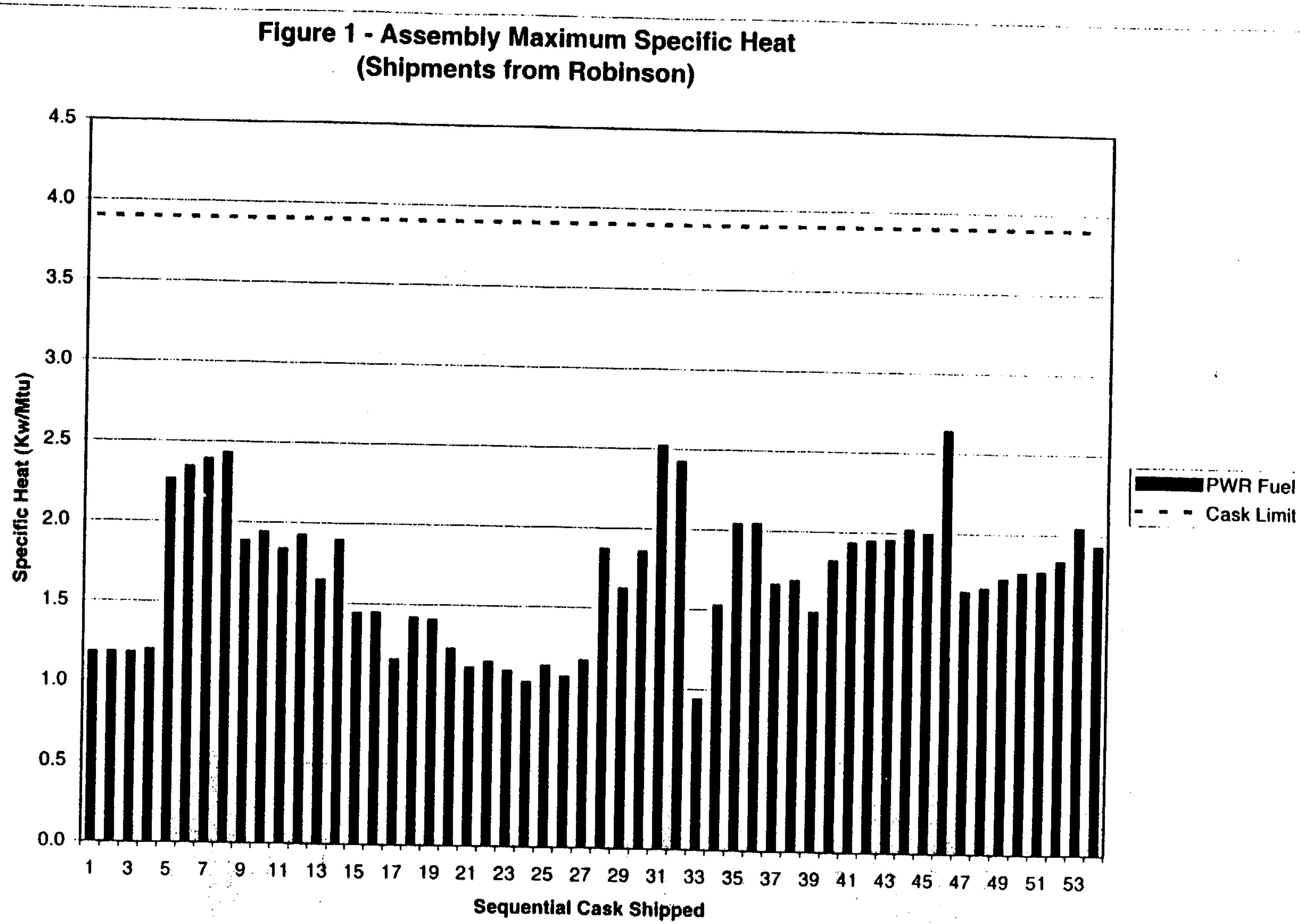
Figure 1 - Assembly Maximum Specific Heat
(Shipments from Robinson)
Specific Heat (Kw/Mtu)
4.5
4.0
3.5
3.0
2.5
2.0
1.5
1.0
0.5
0.0
1 3 5 7 9 11 13 15 17 19 21 23 25 27 29 31 33 35 37 39 41 43 45 47 49 51 53
Sequential Cask Shipped
PWR Fuel
Cask Limit

001189

Figure 2 - Assembly Maximum Specific Heat
(Shipments from Brunswick)

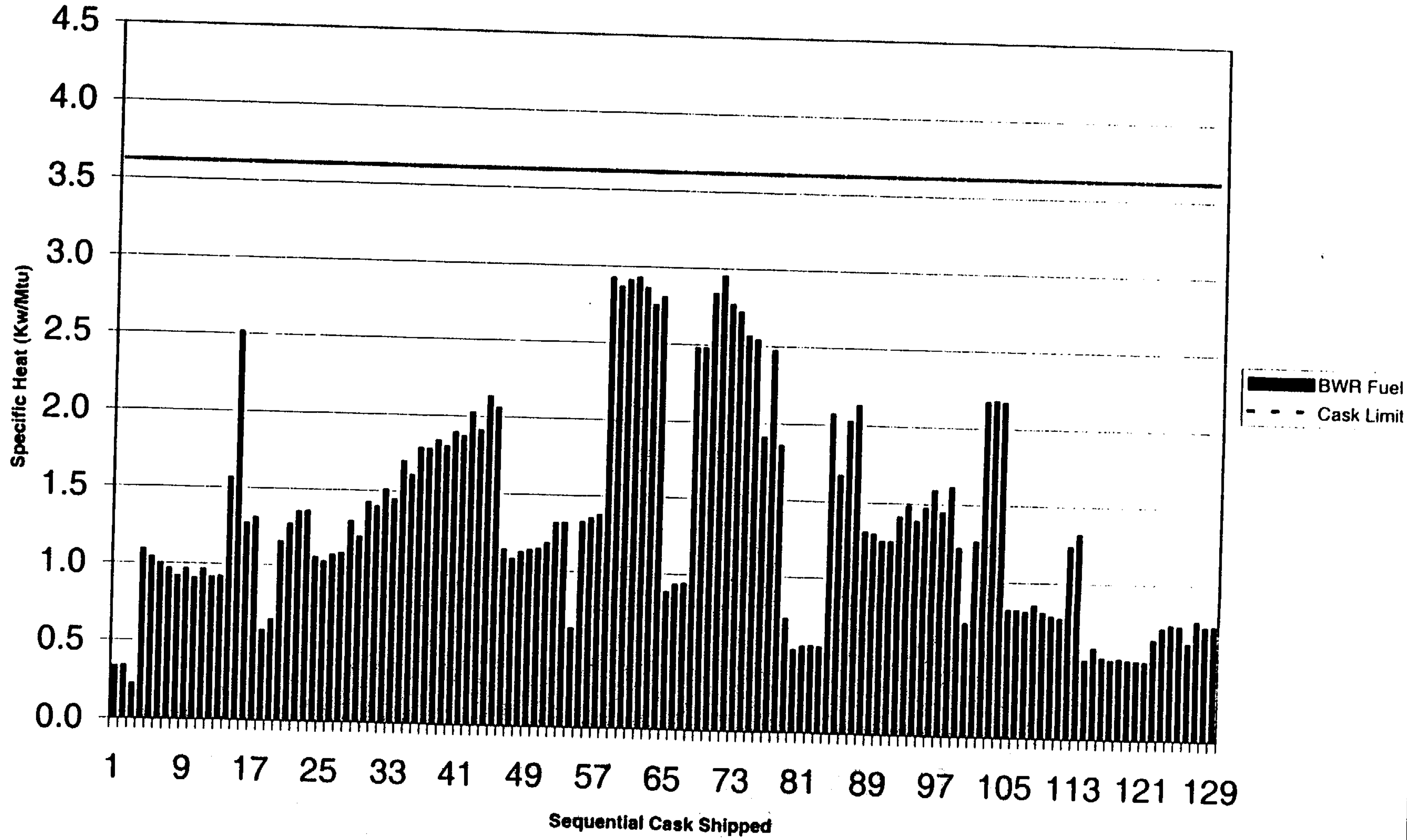

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Figure 3 - Assembly Specific Heat
(PWR Fuel in Harris Pools A and B)

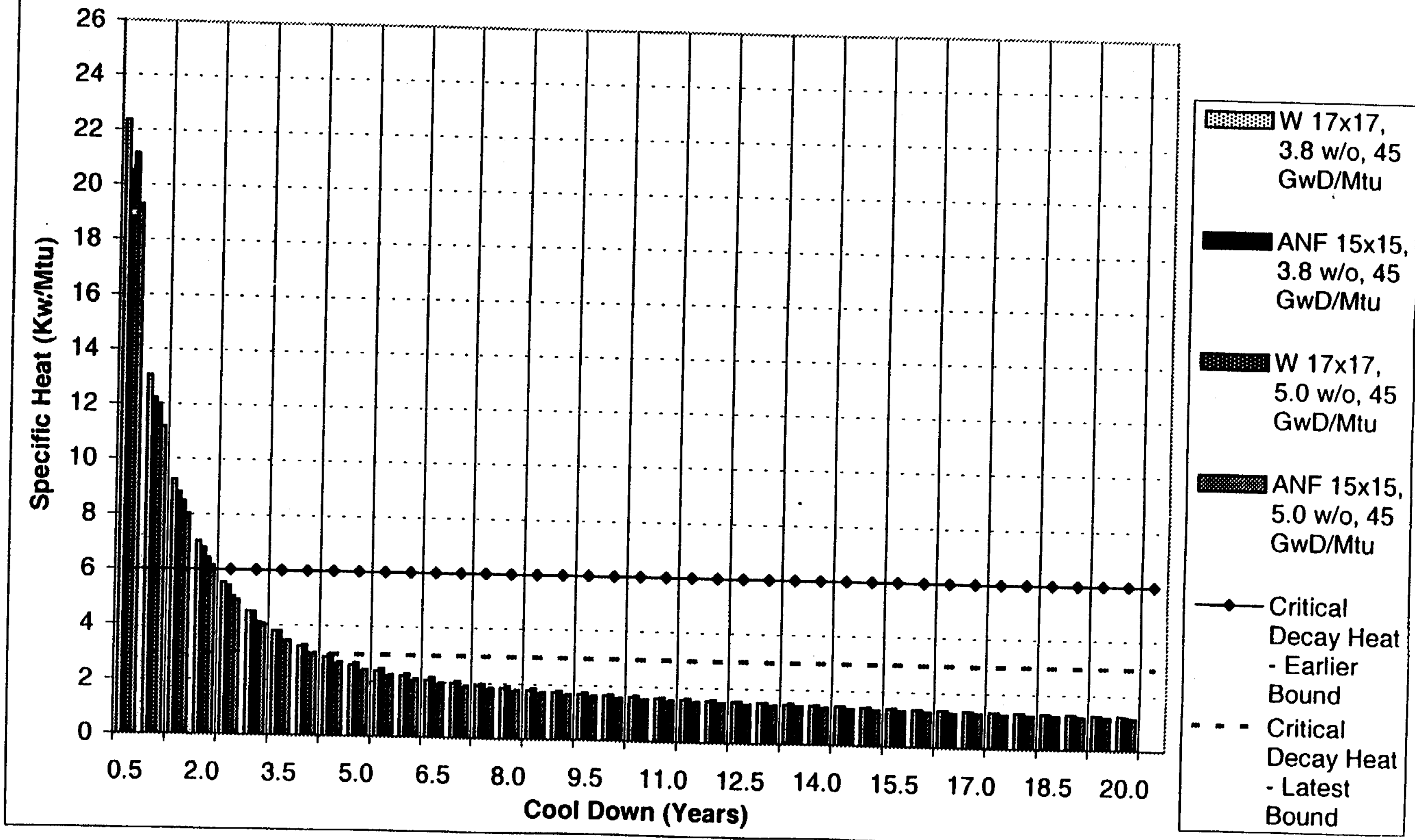

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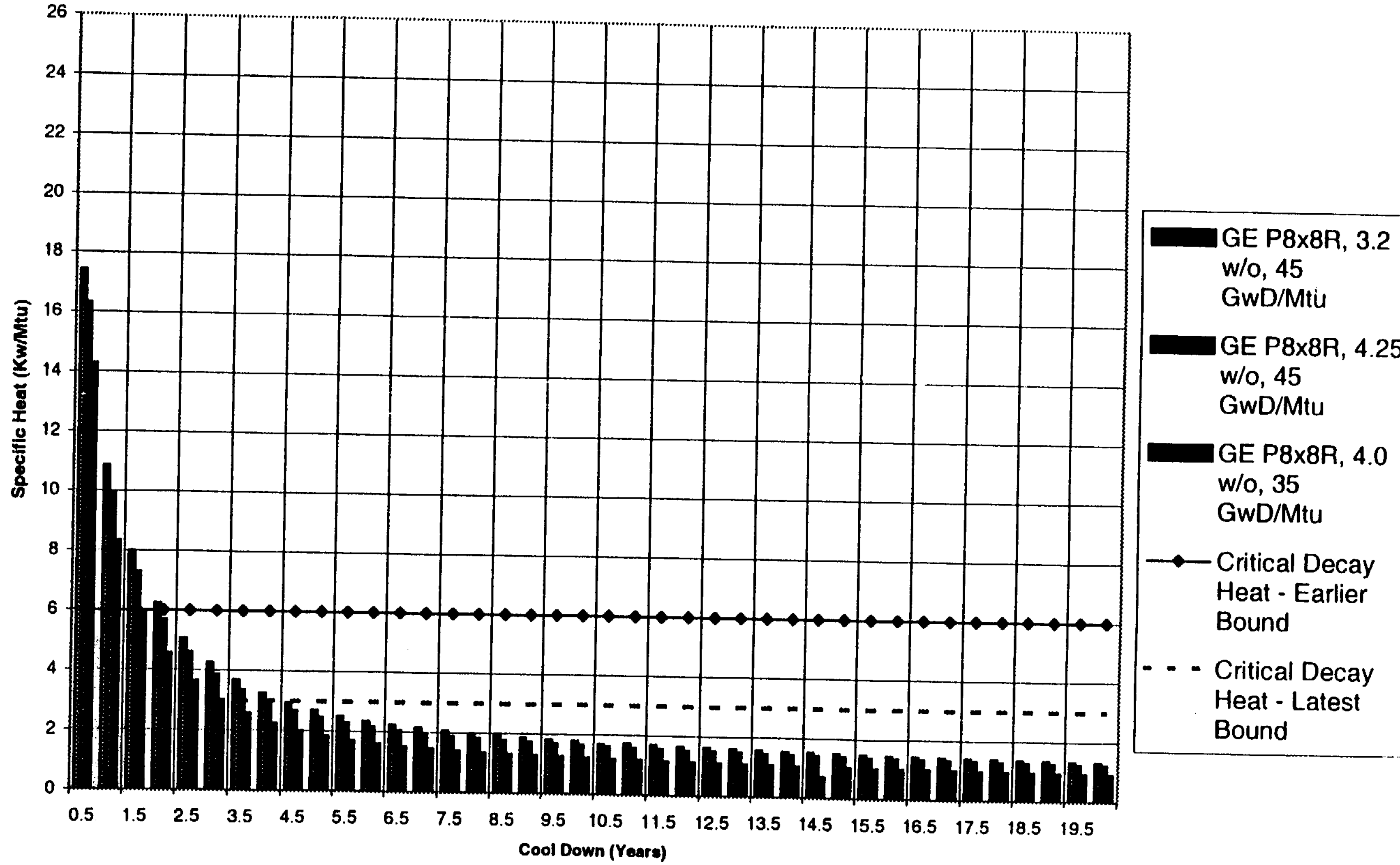
Figure 4 - Assembly Specific Heat
(BWR Fuel in Harris Pools)



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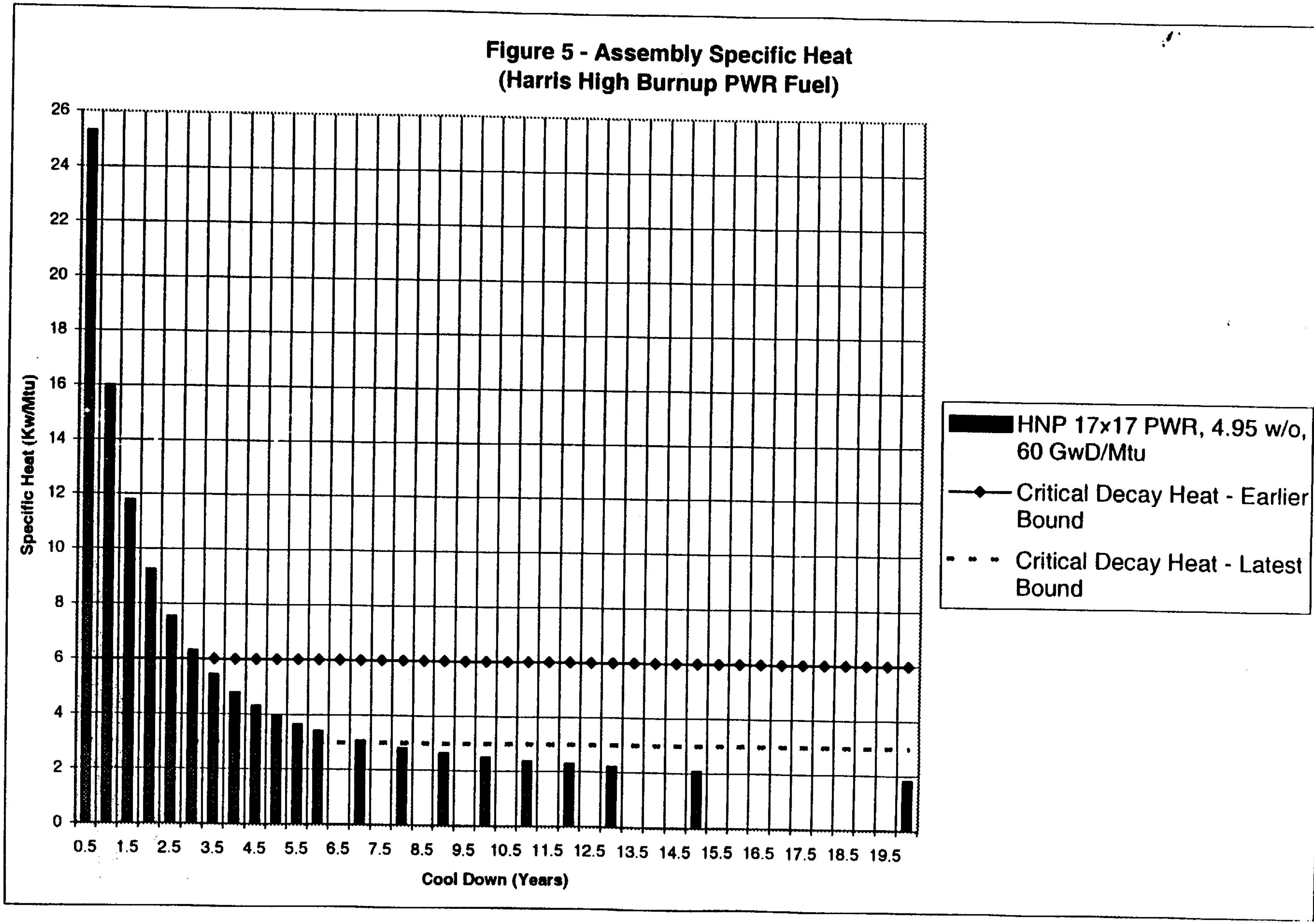

Figure 5 - Assembly Specific Heat
(Harris High Burnup PWR Fuel)
Specific Heat (Kw/Mtu)
26
24
22
20
18
16
14
12
10
8
6
4
2
0
0.5 1.5 2.5 3.5 4.5 5.5 6.5 7.5 8.5 9.5 10.5 11.5 12.5 13.5 14.5 15.5 16.5 17.5 18.5 19.5
Cool Down (Years)
HNP 17x17 PWR, 4.95 w/o, 60 GwD/Mtu
Critical Decay Heat - Earlier Bound
Critical Decay Heat - Latest Bound

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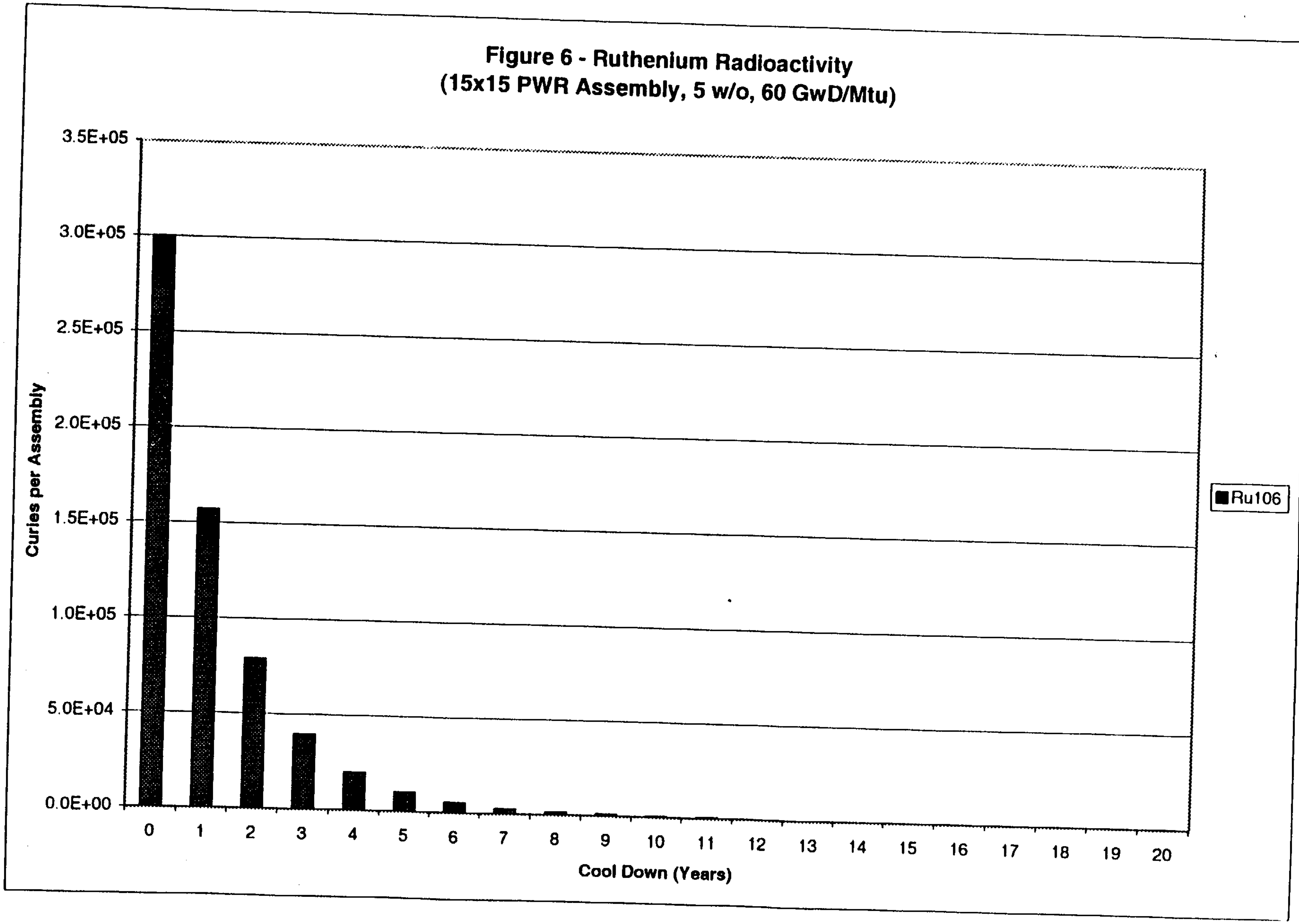

Figure 6 - Ruthenium Radioactivity
(15x15 PWR Assembly, 5 w/o, 60 GwD/Mtu)

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Attachment J

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

2. Susquehanna Safety Evaluation Report.

3. D.A. Lochbaum , 'Comments on the Draft Susquehanna Safety Evaluation Regarding Spent Fuel Pool Cooling Issues, letter to Selin (NRC), November 29, 1994.

4. Taylor, "Resolution of Spent Fuel Pool Action Plan Issues", Memorandum to Commissioners, July, 26, 1996.

5. Callan, "Follow-up Activities on Spent Fuel Pool Action Plan", Memorandum to Commissioners, September 30, 1997.

6. J. Ibarra, W. Jones, G. Lanik, H. Ornstein, S. Pullani, "Assessment of Spent Fuel Pool Cooling", AEOD/S96-02, September, 1996.

7. D. Jackson, T. Eaton, G. Hubbard, "Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants", Draft, February, 2000.

8. W. D. Travers memo to The Commissioners, "Improving Decommissioning Regulations for Nuclear Power Plants", SECY-99-168, June 30, 1999.

9. SRM responding to SECY-99-168, December 21, 1999.

10. D. A. Powers, "re Spent Fuel Fires Associated with Decommissioning", letter to Travers (EDO), November 12, 1999.

11. W. D. Travers. "re Spent Fuel Fires Associated with Decommissioning". December 16, 1999.

12. R. J. Travis. R. E. Davis, E. J. Grove, M. A. Azarm. "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants", NUREG/CR-6451, August, 1997.

13. N. Siu, S. Khericha, S. Conroy, S. Beck, H. Blackman, "Loss of Spent Fuel Pool Cooling PRA: Model and Results", INEL-96/0334, September, 1996.

14. J. Ibarra, W. Jones, G. Lanik, "Operating Experience Feedback Report: Assessment of Spent Fuel Cooling", NUREG-1275 Vol. 12, 1997.

15. J.H. Jo, P.F. Rose, S.D. Unwin, V.L. Sailor, K.R. Perkins, A.G. Tingle, "Value/Impact Analysis of Accident Preventative and Mitigative Options for Spent Fuel Pools", NUREG/CR-5281, 1989.

16. P. Prassinos, C. Kimura, D. McCallen, R. Murray, M. Ravindra, R. Campbell, P. Hashimoto, A. Nafday, W. Tong "Seismic Failure and Cask Drop Analysis of Spent Fuel Pools at Two Representative Nuclear Power Plants", NUREG/CR-5176, January 1989.

17. V. L. Sailor, K. R. Perkins, J. R. Weeks, and H. R. Connell, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82", NUREG/CR-4982, July, 1987.

18. E.T. Burns, D. E. True, V. M. Andersen, "A Review of Draft NRC Staff Report: DRAFT Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants", August, 1999.

19. D.A. Powers, "Draft Final Technical Study of Spent Fuel Pool Accidents Risk at Decommissioning Nuclear Power Plants", letter to R. A. Meserve (NRC), April 13, 2000.

EXHIBIT 3

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

AFFIDAVIT OF STEVEN A. LAUR

COUNTY OF WAKE	)	
	)	ss:
STATE OF NORTH CAROLINA	)	

I, Steven A. Laur, being sworn, do on oath depose and say:

1. I am a resident of the State of North Carolina. I am employed by Carolina Power & Light Company ("CP&L") and presently serve as Superintendent of the Probabilistic Safety Assessment Unit. This unit is responsible to update and maintain plant-specific risk assessment models in a quality manner; maintain consistency in selection and use of risk-based tools across the Nuclear Generation Group; and effectively use risk-based tools to achieve goals in the safety, production, cost and plant license renewal areas. I have as direct reports engineers located at the General Office and at the Brunswick, Harris and Robinson Nuclear Plants. My business address is 410 South Wilmington Street, Raleigh, NC, 27601.

2. I was graduated from University of Central Florida in 1972 with a Bachelor of Science in Mathematics and from Campbell University in 1985 with a Masters, Business Administration. Since receiving my Bachelor's degree, I have been employed by the United State's Navy, Ford Motor Company, RCA Service Company, Florida Power and Light Company, TENERA L.P., and CP&L. During my tenure at CP&L, specific positions held include Project Engineer in the Nuclear Licensing Section, Corporate Nuclear Safety Section, Risk Assessment Unit, and Probabilistic Safety Assessment Unit. I held a Senior Reactor Operator's license at St. Lucie Nuclear Plant, and have diverse technical and management experience in the areas of project engineering, plant operations, safety analysis, risk management, incident investigation and root cause determination. I am a registered Professional Engineer in the State of Florida. My resume is provided as Attachment A to this affidavit.

3. The purpose of this affidavit is to describe the information that was developed by CP&L for use by ERIN Engineering and Research, Inc. ("ERIN") for their performance of an analysis of the sequence of events set forth in the Atomic Safety and Licensing Board's Memorandum and Order dated August 7, 2000 ("Order"). <u>First,</u> I describe the review history of the documents that were used to develop the information. <u>Second,</u> I discuss the specific steps that were taken to ensure that the ERIN analysis was consistent with the plant-specific attributes that were important to the analysis. <u>Finally,</u> I present my conclusions on the quality of the ERIN analyses and the appropriate use of Harris-specific information.

4. My role in the preparing a response to the questions contained in the Board's Order was to provide the updated Harris Individual Plant Examination ("IPE") PSA model and the Harris Individual Plant Examination of External Events ("IPEEE") analysis to ERIN, which they used as the starting point for their analysis of the sequence of seven events set forth on page 13 of the Board's Order. Consistent with CP&L standards for configuration control, I ensured that Harris plant-specific PSA information was provided to ERIN as requested to support the analysis. I was also responsible for ensuring that the resulting ERIN analysis utilized correct and appropriate Harris plant information.

HISTORY OF THE HARRIS PROBABILISTIC SAFETY ASSESSMENT

5. The updated Harris IPE PSA model is a probabilistic safety assessment model that was originally developed for the Harris IPE pursuant to Generic Letter 88-20. The updated Harris IPE PSA model includes: 1) Event trees that model core damage accident sequences and containment response following a core damage event; 2) Fault trees that represent plant systems and failure modes; 3) Initiating event, component failure, and human reliability data; and 4) Special analyses, such as internal flooding and Interfacing System Loss of Coolant Accident ("ISLOCA"). The updated Harris IPE PSA model considers internal initiating events (except internal fires) and applies when the reactor is critical (Modes 1 and 2). The results of the updated Harris IPE PSA model include an estimated annualized core damage frequency. The Harris IPEEE analysis was performed pursuant to

Generic Letter 88-20 Supplement 4. The IPEEE considered 1) seismic risk, 2) internal fire risk, and 3) risk from other external events (e.g., high winds, tornadoes, and nearby facility accidents).

6. The updated Harris IPE PSA and the Harris IPEEE have been reviewed by organizations outside CP&L to ensure they possess a level of quality commensurate with their intended use. An Independent Peer Review of the model was commissioned. This review utilized industry standard guidance for review of PSA models (NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guide") to determine the acceptability of the PSA model for use in analyzing the sequence of events proposed by the Board. ERIN, which was not involved in the development or update of the Harris updated IPE PSA model, performed the review, which is provided as Attachment B. This review provided high marks for the updated Harris IPE PSA model and concluded that, with one exception, the model formed an acceptable starting point for the sequence of events to be analyzed. This exception was the ISLOCA, which is important to the initiating event analysis of the Board's sequence of events, which ERIN determined to be overly conservative. Because the Board required a best-estimate analysis of the sequence of events, I directed ERIN to update the ISLOCA analysis to more realistically model that initiating event. Attachment C contains a table summarizing these reviews.

7. The CP&L management standard is that the Harris updated IPE PSA model be updated and maintained in a quality manner that is equivalent to applicable

portions of 10 C.F.R. Part 50, Appendix B. A CP&L Nuclear Generating Group Common ("NGGC") procedure controls the update and maintenance of the Harris updated IPE PSA model. This procedure requires that: 1) Software used to perform PSA applications and analyses must be qualified and controlled in accordance with NGGC procedures; 2) The PSA models and applications must be described and controlled in accordance with NGGC procedures; 3) PSA model changes, applications and analyses must be performed in accordance with NGGC procedures; and, 4) Errors identified in PSA models, PSA software, PSA methods, or PSA applications must be documented in the CP&L Corrective Action Program.

8. ERIN used the Harris updated IPE PSA and the Harris IPEEE analysis that I supplied as key inputs to their analysis of the Board's sequence of events. Use of these Harris models and analyses ensured that the ERIN analysis was based on the latest available, plant-specific risk information.

HARRIS SPECIFIC INFORMATION

9. To ensure that the ERIN analysis of the Board's sequence of events was consistent with the plant-specific attributes important to the analysis, I coordinated a review of the report by a multi-disciplinary team of individuals within CP&L who have knowledge and expertise in one or more of the areas covered by the report. This review included ensuring that the inputs provided by CP&L were correctly incorporated into the analysis. The review also validated that the methodology

used by ERIN was appropriate for use at Harris. The review confirmed that assumptions and statements made in the report concerning plant-specific features and characteristics were appropriate. I reviewed portions of the analysis and reviewed the comments provided by the other members of the CP&L review team.

10. When the Board's sequence of events was promulgated, I recommended an outside contractor be retained to perform the analysis and selected ERIN. I directed ERIN to provide a best-estimate risk assessment of the sequence of events described in the Board's Order and to report the result in terms of an estimate of the annual frequency of the entire scenario. This analysis was to include not only internal events as modeled in the Harris updated IPE PSA model, but also sensitivity analyses of the scenario frequency to other initiating events, including dominant internal fires and seismic events. The analysis was also to consider the sensitivity of the results to core damage events during shutdown conditions.

11. In order to support the analysis by ERIN, CP&L provided the plant-specific information specified in the table provided as Attachment D to this affidavit. Cognizant CP&L personnel reviewed each submittal to ERIN to ensure that accurate information was provided.

12. ERIN personnel toured the Harris facility on two occasions to gather plant-specific data for the analysis of the Board's sequence of events. ERIN personnel discussed pertinent plant information with cognizant Harris personnel, including

sources of make-up water to the spent fuel pools and accessibility of the reactor auxiliary building and fuel handling building during accident conditions.

CONCLUSION

13. In conclusion, the ERIN analysis of the sequence of events postulated in the Board's Order used appropriate Harris plant-specific information. The ERIN analysis utilized the information contained in, and builds upon, the existing Harris updated IPE PSA model and Harris IPEEE analysis, which are plant-specific studies. The Harris updated IPE PSA model was independently peer reviewed and found to be of high quality and appropriate for the analysis of the Board's sequence of events. The CP&L review of the ERIN analysis of the Board's sequence of events confirmed that the assumptions and data used by ERIN reflect the relevant attributes of the Harris plant.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2000.

Steven A. Laur
Steven A. Laur

Subscribed and sworn to before me
this 15 day of November 2000.

Wanda C Langston

My Commission expires: 9-15-02

WANDA C. LANGSTON
NOTARY
***
PUBLIC
WAKE COUNTY, N.C.

STEVEN A. LAUR, P.E.

Project Engineer

Professional Qualifications

Mr. Laur has over 20 years of engineering and managerial experience, which includes extensive experience in the operation, training, quality assurance, maintenance, and supervision of nuclear power reactors. He held a Senior Reactor Operator license at St. Lucie Nuclear Plant, and has diverse technical and management experience in the areas of project engineering, plant operations, safety analysis, risk management, incident investigation and root cause determination.

He has provided management consulting to assist nuclear plants in determining the root causes of and developing effective solutions to their performance problems. Additionally, Mr. Laur conducted performance-based assessments of nuclear utilities, with focus on operational readiness, independent oversight, quality verification, and regulatory compliance issues.

Mr. Laur was Project Engineer assigned to the Nuclear Licensing Section during the licensing process for Shearon Harris Nuclear Power Plant. He supervised four Senior Engineers and Specialists and seven Technical Aides, Licensing Assistants, and Production Assistants. He was the Harris Licensing Engineer during the ASLB hearings, receipt of the Final SER, and the ACRS sub-committee and full committee hearings for the plant. He served on the H. B. Robinson trip reduction team and conducted special investigations of plant events at all four CP&L nuclear units to determine root causes and assess the adequacy of corrective actions. He conducted internal assessments of CP&L's nuclear sites patterned after the NRC SALP program. Mr. Laur performed independent reviews as a member of the standing nuclear safety organization and provided insight to the organization, charter development, and startup phase of the CP&L Nuclear Assessment Department.

He is currently employed by Carolina Power & Light Company as the supervisor in charge of the Probabilistic Safety Assessment Unit, which is responsible for Risk analyses for the Brunswick, Harris, and Robinson nuclear plants.

Education/Special Training

MBA	Campbell University, 1985
B.S.	Mathematics, University of Central Florida, 1972

U.S. Navy Nuclear Power Training, Bainbridge, MD and West Milton, NY, 1975-1976

EG&G Accident Investigation Course, including Management Oversight and Risk Tree (MORT), 1987 (1 week)

TENERA Integrated Risk Management System training, October, 1994

Experience

1998 - Present

Superintendent – PSA Unit, Carolina Power & Light Company. First-line supervisory position in charge of Unit responsible to update and maintain plant-specific risk models in a quality manner (equivalent to applicable portions of the 10CFR50 Appendix B program): maintain consistency in selection and use of risk-based tools across NGG; and effectively use risk-based tools to achieve goals in the safety, production, cost and plant life extension areas. Position has as direct reports engineers located at the General Office and at Brunswick, Harris and Robinson sites.

1995 - 1998

Project Engineer, Carolina Power & Light Company. Currently provides PRA support to the Brunswick, Shearon Harris, and H. B. Robinson nuclear plants. This support includes model development and maintenance as well as applications of the PRA to facilitate managing the risks associated with nuclear power operations. Familiar with current PRA methods and practices, including fault tree and event tree analysis, data gathering and analysis, human reliability analysis, and quantification of PRA models using the CAFTA computer code. Performed specialized PRA application studies to determine critical components for maintenance, to determine the risk associated with online maintenance activities, and to provide justification for technical specification changes.

Performed special fault tree model of the Harris emergency diesel generator control system to identify critical components for predictive and preventive maintenance to ensure continued high reliability. Greatly enhanced the quantification methodology for the H. B. Robinson PRA, with the result that the level 1 (core damage) model quantification time was reduced from over 24 hours to around 15 minutes. Provided analyses to evaluate the risk associated with several events at the Brunswick plant. Provided statistical analysis of increased control rod scram times for Brunswick unit 1.

1992 - 1995

Senior Consultant, TENERA, L.P. Provided management consulting to assist nuclear plants in determining the root causes of and developing effective solutions to their performance problems.

From December, 1991 through June, 1993, performed the human reliability analysis (HRA) for the Dresden and Quad Cities Stations as part of the Commonwealth Edison Company's Individual Plant Examination (IPE). Utilized the NUREG/CR-1278 (THERP) methodology. Observed licensed operators perform simulator exercises for Dresden and Quad Cities in support of the HRA. Reviewed plant response (event) trees for Dresden and Quad Cities; developed a fault tree model of the contaminated condensate storage tank inventory control function for Quad Cities. Performed an internal review of the HRAs for the Monticello and Clinton PRA models.

He assisted the Palisades plant in preparing for an NRC Diagnostic Evaluation Team (DET) inspection. He facilitated the identification of performance improvement initiatives and facilitated validation workshops to ensure all major problems facing the station were identified. In a similar effort, he assisted the Quad Cities plant in improving their ability to operate the plant efficiently and effectively, including work on the station Management Plan, prioritization, ranking, and management of issues, setting management expectations for site safety culture, and providing guidance regarding how to respond to their NRC Diagnostic Evaluation Team inspection.

In an effort to assist Commonwealth in addressing the placement of the Zion and Dresden plants on the NRC watchlist, assigned the role of technical team leader for the root cause portion of a comprehensive evaluation program (May, 1992). As a follow-on effort, was again the technical team leader for the root cause evaluation of the performance of the Quad Cities station in preparation for and NRC Diagnostic Evaluation Team (DET) inspection (July, 1993).

In 1992, participated in a performance-based assessment of the PSE&G Quality Assurance/Nuclear Safety Review organization. During this project, interfaced with PSE&G site supervisory and management personnel. Conducted technical information-gathering interviews with site personnel at a variety of levels and reviewed a variety of documentation in support of this project including the products produced by the QA/NSR organization, LERs, incident reports, QA reports, INPO evaluation reports, and NRC correspondence. Primary author of the resultant report to the client.

As a direct result of the 1992 assessment, awarded sole-source project to perform follow-on work for the PSE&G Nuclear Safety Review group.

In February, 1993, taught five one-day classes on the subject of NRC reporting requirements at both the PG&E corporate offices in San Francisco and at the Diablo Canyon nuclear plant. In a separate effort for PG&E, developed a technical basis document to support sampling plans for samples from a finite population without replacement (based on the hypergeometric distribution). Derived algorithm and developed corresponding computer code to aid in the calculation of the cumulative hypergeometric distribution.

1983 - 1991 Carolina Power and Light Company:

1991 *Project Engineer*. Risk Assessment Unit. Provided analysis and evaluation of nuclear power plant design features utilizing PRA techniques. including work necessary to respond to the NRC IPE requirements and severe accident policy. Provided human reliability analyses to support plant-specific probabilistic risk assessments. Utilized fault tree models and core damage sequences to provide comparative analyses of options for management decision making regarding modifications. justifications for continued operation. and requests for NRC waiver of compliance issues.

Developed fault trees to support a Brunswick plant application: Comparison of various proposed changes to the onsite electric supply and distribution system. Quantified the existing Brunswick PRA model using CAFTA computer code to determine the core damage frequency impact for the above application. Reviewed fault trees and event trees for the H. B. Robinson plant as part of performance of the human reliability analysis (HRA). Utilized the EPRI time reliability correlation. the EPRI decision tree. the NUREG/CR-1278 (THERP). and the NUREG/CR-4772 (ASEP) methodologies. Observed licensed operators perform simulator exercises for Robinson and Brunswick.

Provided statistical expertise in the development of sampling plans at the Brunswick plant. Wrote a computer program and performed Monte-Carlo simulation of emergency diesel generator starts.

1985 - 1990 *Project Engineer*, Corporate Nuclear Safety Section. Provided independent review of plant safety analyses and documents to assure proper maintenance of nuclear safety; developed recommendations for improvements in nuclear safety; assured proper corrective action taken to prevent recurrence of events involving nuclear safety; and conducted special evaluations of selected safety-related matters.

1983 - 1984 *Project Engineer*, Nuclear Licensing Section. Supervised up to 11 engineers and technicians within Licensing and functioned as first line management for projects and programs under his cognizance. Responsible for development and implementation of a computerized system for centrally tracking the company's regulatory commitments.

1981 - 1982 *Technical Staff Engineer*, Florida Power and Light Company St. Lucie Plant. Acted as licensing coordinator between the plant departments and the general office. Responsible for performing Technical Specification Surveillance Testing and for secondary plant performance testing and evaluation. Successfully completed training programs covering plant systems with emphasis on mitigating core damage in the event of a nuclear accident; qualified to stand watch as needed in the capacity of Shift Technical Advisor. Received Senior Reactor Operator license from the NRC.

1979 - 1981 *Project Engineer/Trial Coordinator*, RCA Service Company. Supported submarine acoustic trials at the Atlantic Undersea Test and Evaluation Center.

1978 - 1979 *Design Engineer*, Ford Motor Company. Engineer in the C-4 automatic transmission design unit.

1974 - 1978 *Nuclear Submarine Service*, U.S. Navy. Served in various engineering assignments aboard a nuclear submarine.

Special Qualifications

Registered Professional Engineer
Senior Reactor Operator License, St. Lucie Nuclear Plant
Nuclear Submarine Engineering Officer of the Watch

Professional Affiliations

Member of the American Nuclear Society (ANS)
Member of the Institute of Electrical and Electronics Engineers (IEEE)

PEER REVIEW OF SHEARON HARRIS PSA

Prepared for:

Carolina Power and Light Company

Prepared by:

ERIN® Engineering and Research, Inc.
2111 Palomar Airport Road, Suite 180
Carlsbad, CA 92009-1419
(760) 929-0870

November 7, 2000
Final Report

001210

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LIST OF TABLES

SECTION 1
INTRODUCTION

1.1 SCOPE

The purpose of this report is to document the results of an independent review of the Shearon Harris PSA ("Harris PSA") that was performed in October 2000 by ERIN Engineering and Research, Inc. under contract with Carolina Power and Light Co. The purpose of the review is to support the use of the Harris PSA in an evaluation of scenarios that could cause or exacerbate a loss of cooling to the spent fuel pool at the Harris plant. The objectives of the review are to assess the quality, scope, and technical adequacy of the existing PSA models to support PSA applications and specifically the evaluation of postulated spent fuel pool scenarios. In addition, the peer review determines what enhancements to the PSA models are desirable to complete the spent fuel pool evaluation.

The scope of the peer review was limited to a documentation review. No onsite visit or direct interaction with the PSA was performed as part of this peer review.

1.2 REVIEW APPROACH

The approach to conducting the review is based on a PSA peer review process that was originally developed by the Boiling Water Reactor Owners Group and is now being used by all of the existing LWR owners groups, including the WOG (Reference [1]).

The purpose of this report is to document a peer review of the Harris PSA using the methodology and checklists developed for WOG peer reviews in order to support the application to the spent fuel pool evaluation.

The PSA Peer Review Process described in Reference [1] provides a structured process to review a PSA using a checklist that examines the PSA in terms of 11 elements and 209 sub-elements. These PSA elements are listed in Table 1-1.

Table 1-1

PSA Elements

Element Code	PSA Element
IE	Initiating Events
AS	Accident Sequence Evaluation
TH	Thermal Hydraulic Analysis
SY	System Analysis
DA	Data Analysis
HR	Human Reliability Analysis
DE	Dependencies
ST	Structural Response
QU	Quantification
L2	Containment Performance
MU	Maintenance and Update Process

Each of the above elements is further broken down into a total of about 209 sub-elements to permit a structured and detailed examination of the PSA and its associated models and documentation.

A summary of the major review findings and recommendations for future updates is provided in Section 3.

1.3 REVIEW TEAM

A review team consisting of Karl Fleming, Tom Daniels, Jeff Gabor, Ed Burns, and Grant Tinsley carried out this review. Karl Fleming is a recognized expert in developing and applying PSA technology and in performance of PSA peer reviews. He served as the leader of the review team and was responsible for the review of about half of the PSA elements and for the preparation of this report. Ed Burns was the principal author of the PSA Peer Review process that was used in this report and has been involved in more than 25 peer reviews that have been completed using this methodology. Mr. Fleming and Dr. Burns are also principal authors of the ASME PRA standard (Reference [2]) which is currently under development to support the PSA review process. Mr. Daniels has extensive experience as a PSA practitioner and is also playing a lead role in modifying the existing Harris PSA models to support the evaluation requested by the Atomic Safety Licensing Board. Jeff Gabor is a recognized expert in PSA technology and was a principal author of the accident progression analysis methodology that was used in the Harris PSA to support success criteria, sequence timing and source term assessment. Grant Tinsley also has extensive experience as a PSA practitioner and has experience in applying the industry peer review process at several plants. As a team, this group is adequately qualified and experienced to reach technically sound conclusions about the technical quality of the PSA and its capability to support the spent fuel pool licensing evaluation.

As a final note, while a grading system is used to provide a reasonable degree of consistency in the peer review process, the most valuable result of this type of review is a focused set of strengths and weaknesses that the PSA group can use in existing applications. Hence, the grades themselves do not assure PSA quality, but rather the supporting strengths and weaknesses that are identified from the grading process provide a roadmap for steps that should be taken to implement quality PSA applications.

SECTION 2
REVIEW RESULTS FOR PSA ELEMENTS

The technical review was organized into the PSA elements listed in Table 1-1 so that the Industry PSA Peer Review Process could be closely followed. The results of the review for each element are provided in Sections 2.1 through 2.11 below.

2.1 IE - INITIATING EVENTS

The scope of this element includes identification and grouping of initiating events (IE) and the estimation of initiating event frequencies. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-1. The key points from this evaluation are summarized below:

- The extent of updating of the initiating events analysis and the use of formal calculations to document the in-house reviews provides good evidence that the initiating events analysis reflects the as-built and as-operated plant.
- The selection of an initiator to represent loss of normally running and standby charging pumps is unusual, but reflects thoughtful consideration of plant specific and unique features; this event leaves the plant with no high pressure makeup if the scenario should develop into a LOCA.
- The statistical methods used to estimate the initiating event frequencies include both classical (chi-squared) and Bayesian. In future updates all events should be quantified in a Bayesian framework for consistency, however this does not have an appreciable impact on the numerical results.
- The analysis of the frequency and recovery of loss of offsite power reflects the state of the art in the early 90's and would benefit from more recently collected and analyzed data such as those in Reference [3]. An updated analysis would likely show that the current analysis is somewhat conservative.

- The IE frequency analysis is well documented and is traceable to sources for generic and plant specific evidence, but the sources should be updated and a consistent Bayesian methodology should be used for all events.
- The steam and feed line break IE frequencies are well documented but based on somewhat old references and should be updated. An updated analysis would likely show that the current analysis is somewhat conservative.
- Some of the IE frequencies developed from failure rates are not weighted by a plant availability factor and are therefore biased on the high side by the inverse of the availability factor. An updated analysis could show that the current analysis is somewhat conservative by 20% to 30%.
- A check of the Bayes update calculations is included as part of the data evaluation in Section 2.5
- The report contains a very good discussion of how initiating events are binned based on functional categorization and binning trees.
- ISLOCA initiating frequency evaluation appears quite conservative compared with other similar plants. This could be reassessed to make the PSA more realistic. Specific comments on the ISLOCA models are included with the review of the Level 2 PSA element in Section 2.10.

The Harris PSA report text says that IE fault trees for certain systemic events such as service water were linked with mitigation fault trees. These initiating event fault trees were reviewed indicating a number of issues that are listed below.

- The fault tree models used for initiating event frequencies include combinations of two or more basic events with an annual exposure time. An example is a cutset at 3.1E-08 involving a Loss of Instrument Air initiator (%T13) that includes the basic events ACP1ANS%FN and ACP1BN%SFN representing the A and B compressors. Both of these basic events use the compressor failure rate times 8760 hours. Common cause failures of 2 and 3 compressors are included separately. The fault tree logic should be such that these combinations are not generated or the combinations can be included in the mutually exclusive file. The exposure time for the first failure is 8760 hours, but the exposure time for any subsequent failures is the repair time for the first component failed. Similar combinations exist in the Loss of Normal Service Water (%T9) and Loss of CCW (%T11) initiating event models.

- The fault tree model for the Loss of CCW initiating event does not include combinations of pump failure to run with a failure to reseat of the failed pump's discharge check valve. This is also true for the Loss of Normal Service Water (%T9) and Loss of Instrument Air (%T13) initiating events.

- The models for loss of a 6.9kV bus initiating event consist of a single basic event representing bus failure during operation multiplied by 8760 hours. The model should account for other failure modes, such as circuit breaker transfers open while indicating closed.

- The model for loss of non-safety DC bus DP-1A (%T15) consists of 2 basic events, one for battery short and one for distribution panel failure during operation. The model should account for other failure modes, such as battery charger failure during operation in combination with battery failure on demand, assuming the battery is sufficient to supply loads without the charger.

- The inclusion of "failure of NSW pump strainer to run" leading directly to Loss of Normal Service Water (%T9) may be overly conservative. It is a dominant contributor to the initiating event frequency.

- The fault tree logic correctly treats the dependencies associated with some of the same components and failure modes appearing in the initiating event frequency fault trees and the mitigation function fault trees by using different basic event names for each. Flags are used to make sure that equipment is not permitted to fail more than once.

- The initiating events quantified using fault trees as well as those quantified using data did not have an availability factor applied to account for the probability that the plant is in operation at the time of the event; this creates a conservative bias on the order of 10% to 20% in the initiating event frequencies.

- In the loss of charging system initiating event model, it is not clear that the correct times were used in modeling cutsets with valves transferred closed in combination with pump failures.

Table 2-1

Evaluation of Initiating Events Elements

Attribute	Assessment
Guidance	It is not known whether CP&L has any separate guidance documents but the PSA documentation of the initiating event analysis is very thorough and hence, serves as useful guidance to support PSA updates.
Grouping	The identification and grouping of initiating events was very thorough and comprehensive. Good use of insights from EPRI ATWS study and PSAs on other Westinghouse PWR plants. Event trees used for binning provide a very clear description of how each event was treated.
Treatment of Support System/Special Initiators	A systematic search for plant specific important initiators was performed in the system notebooks. The final list of initiating events includes a relatively large number of support system initiators which is appropriate for PWR plants. The only PSA in the country that has a larger list of initiating events is probably Calvert Cliffs, which is an industry outlier in this respect. The Harris PSA list appears to be as complete as other PSAs such as South Texas which are regarded as quality PSAs. A very good level of completeness could be brought to excellent if the list was expanded to consider common cause bus failures as initiating events (there have been a few events at other plants where degraded components inside switchgear cabinets have failed energetically , and other events in which fuses inside multiple inverters have blown taking out multiple buses at the same time. Another category of events are failures in multiple support systems that functionally interact, such as failure of a CCW train with a SW train supporting the opposite CCW train (some refer to the occurrence of initiating precursors). The ISLOCA frequency appears unrealistically high and may bias the results of the PSA in certain applications.
Data	One aspect of the initiating events analysis that has significant room for improvement. The initiating event frequency calculations were very clearly documented and traceable back to the source data, but several problems were noted. One is the inconsistent use of classical and bayesian statistical method. The other is the need to update the generic database as the NRC via INEEL has recently published some more up to date and more realistic sources. Such an update is expected to result in somewhat lower initiating event frequencies for some events.
Documentation	The documentation of the initiating events analysis is excellent and would likely get a grade level 4. The documentation is very clear and makes the analysis quite transparent. There is very good use of figures and tables and key data and assumptions are justified and the bases traceable back to the source.
Recommended Enhancements	Update the generic data inputs from the INEEL study and perform a consistent Bayes' update for all initiating events using the generic distributions from the INEEL study. Update the loss of offsite power frequency and time to repair distributions. Include the plant availability factor in all initiating event models.
Overall Process Assessment	The initiating events analysis is capable of supporting risk significance determinations with deterministic input. The current initiating events analysis would be peer reviewed at a solid Grade 3, and if the recommended enhancements were made with the same level of quality of other aspects of the analysis, this element would be at Grade 4.

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Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- ■ **Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input**
- ☐ Grade 4 - Provides Primary Basis For Application

2.2 AS – ACCIDENT SEQUENCE EVALUATION

The scope of this element includes identification and grouping of functional accident sequence categories, event tree development, core damage and plant damage state characterization and interface with other elements such as initiating event identification and systems dependency analysis. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-2. The key points from this evaluation are summarized below:

- Sequence definition is accomplished via event sequence diagrams and event trees that are very clearly documented
- Very thorough treatment of transient induced LOCAs via PORV and safety valve pressure challenges.
- Used NUREG-1150 RCP seal LOCA model; according to the Westinghouse Owners' Group, the assumption of a 1.5 hour time to begin leaking is now regarded as questionable for some of the RCP seal failure modes, especially the so-called popping failure mode. A more up to date model, such as the Rhodes model, should be considered for the next update.
- Good development of event tree logic from safety functions and clear presentation of success criteria.
- Transient ESD/ET does not credit alternate feedwater as indicated in the EOPs/FRGs for sequences in which there is failure of main feed, auxiliary feedwater, and feed and bleed.

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- Figure 4-5 has no labels, just graphics. (not critical to the review)
- Table 4-6 presents pump capacities but not the pressures that corresponds to these capacities; it would also be helpful know the shutoff heads of these pumps.
- No credit taken for operators to terminate sprays results in a conservative treatment of RWST depletion time.
- The LOSP event tree does not discuss how RCP seal LOCAs are treated, nor does it describe how offsite power recovery is factored in. This needed to be determined from other documents. It is confusing that SBO sequences include both Auxiliary Feedwater System (AF) success and AF failed sequences. There must be logic in the recovery file that effectively creates different sequences. Certain sequences appear to be missing: Loss of offsite power, RCP seal LOCA starts, offsite power successfully restored, core damage due to failure to continue to provide makeup or other LOCA mitigation functions. This would be evidenced by transfers to the Small LOCA event tree if these conditions were met
- Bridge tree is judged to address all relevant plant damage state issues.

Table 2-2

Evaluation of Accident Sequences Element

Attribute	Assessment
Guidance	It is not known whether CP&L has any separate guidance documents but the PSA documentation of the accident sequence analysis is very thorough and hence, serves as useful guidance to support PSA updates.
Success Criteria and Bases	Success criteria tabulated for each event tree and is generally traceable back to the source calculation. A number of the success criteria are acknowledged to be conservative and lead to an overstatement of risk for some applications. However, the success criteria developed from the assumption that RCP seal LOCAs do not initiate for 1.5 hours alludes to an error in NUREG-1550 in how information provided by Westinghouse for the use in the expert elicitation was interpreted. Information from the WOG indicates that some RCP seal failure modes may occur almost immediatey. In addition, the WOG has indicated that if seal injection is lost for 10 minutes it should not be restarted to prevent a thermal shock induced seal failure.
Accident Scenario Evaluation (Event Tree Structure)	There was a good use of safety functions and event sequence diagrams to document sequence development. The event tree structures are simplified in relation to many other plants using the same methodology, but otherwise well documented. The LOSP event tree appears to be overly simplified; for example SBO sequences with successful and unsuccessful auxiliary feedwater are collapsed into the same sequence. It does not appear that LOSP sequences with RCP seal LOCAs and successful OSP recovery have been processed as SLOCA sequences, and this simplification is optimistic. This does not have a significant impact on the baseline CDF but could be a problem for certain applications in which changes to the ECCS system were being evaluated.
Interface with EOPs/AOPs	There are number of references to mitigation possibilities that are covered in the EOPs and FRGs to justify and develop the event trees, however in several of these cases the full mitigation addressed in the EOPs is not credited creating some conservatisms.
Sequence End State Definition/Treatment	The Level 1 end states are success and core damage; a Bridge Tree is used to interface with the Level 2 Containment Analysis through the definition of a set of plant damage states. The plant damage states track sufficient information to resolve dependencies between the Level 1 and Level 2 event sequence models. The Level 1 end state provides a clean interface with the Level 2 anlysis and is clearly described.
Documentation	The documentation is excellent making the event tree analysis and assumptions quite transparent.
Recommended Enhancements	A more up to date RCP seal LOCA model should be used, and for some applications it might be fruitful to eliminate some of the above noted conservatisms. The LOSP event tree should be expanded and the links to the Small LOCA tree for RCP seal LOCAs and successful OSP recovery created and/or identified.
Overall Process Assessment	When the recommended enhancements are incorporated this element of the PSA is of sufficient quality to support risk significant evaluations with deterministic input. However even without these enhancements, this aspect of the PSA is adequate to meet most applications.

Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- © **Grade C3 - Supports Risk Significance Evaluations w/Deterministic Input(conditions)**
- ☐ Grade 4 - Provides Primary Basis For Application

2.3 TH – THERMAL HYDRAULIC ANALYSIS

The scope of this element includes the engineering and thermal hydraulic analysis that supports the PSA in several key areas including success criteria for the accident sequence model and systems analysis, time windows for HRA, room heatup analyses, and plant and containment analyses that support the Level 2 PSA. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-3.

Table 2-3

Evaluation of Thermal Hydraulics Element

Attribute	Assessment
Guidance	It is not known whether CP&L has any separate guidance documents but the PSA documentation of the thermal hydraulics analysis is very thorough and hence, serves as useful guidance to support PSA updates.
Best Estimate Calculations	The thermal hydraulics analysis supports the PSA success criteria, time windows for human operator actions, and analysis to support the Level 2. These analyses include simple mass and energy balances, plant specific MAAP analyses, and operator training simulator exercises for key accident sequences. In general, these analyses are realistic, technically sound, well documented and traceable to adequately support the PSA models and assumptions.
Room Heatup Calculations	Room heatup calculations were performed and appear to be realistic and yield reasonable results.
Documentation	This TH analysis is among the best documented for a PSA that the review team has seen. There are many figures and charts that display a deep understanding of the results and their limitations.
Recommendations for Enhancements	Maintain this quality level as the PSA experiences future updates.
Overall Process Assessment	This element of the PSA is a key strength and is capable of supporting the envisioned risk informed applications.

Recommended Element Grade:

☐ Grade 1 - Supports Assessment of Plant Vulnerabilities

☐ Grade 2 - Supports Risk Ranking Applications

☐ Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input

■ **Grade 4 - Provides Primary Basis For Application**

2.4 SY – SYSTEM ANALYSIS

The scope of this element includes the analysis of systems that support the PSA, including the identification of system functions addressed in the PSA, system success criteria, system dependencies, potential for causing an initiating event, fault tree models that support the accident sequence models, and probability models to support the accident sequence quantification. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-4.

Table 2-4

Evaluation of Systems Analysis Element

Attribute	Assessment
Guidance	It is not known whether CP&L has any separate guidance documents but the Systems Notebooks are very thorough and hence, serves as useful guidance to support PSA updates.
Systems Modeled	23 original systems notebooks and one additional system notebook (SFP cooling & cleanup) have been developed. All expected SSCs (i.e., PWR CDF contributors) modeled. "Extra" SSCs (above usual scope) observed including demin water, non-safety related SW and main FW (and SFP cooling & cleanup as previously mentioned).
System Model Structure (Fault Tree)	The fault tree models are extensive. They are consistant with or exceed industry practice in all aspects. Excellent system-to-system consistency in format, structure, and level of detail.
Success Criteria	Plant-specific T-H calcs for CVCS, AFW, RHR, ESW, MFW. Extensive, appropriate use of plant-specific MAAP model to support systems success criteria development.
Recommended Enhancements	More frequent system updates (last complete update was 1995). Consider more flexible, independent update for those SSCs undergoing significant modifications could be considered. This should be possible given the software either available or imminent (i.e., FORTE solution engine, etc.) and the move with the "2000" update to make all notebooks controlled engineering calculations.
Overall Process Assessment	Very difficult to find fault with even the 1995 notebooks; the few completed "2000" notebooks reviewed were even better. One of the very best examples of systems analysis and documentation this reviewer has seen in the industry, including those studies in which he has managed or participated. The system-to-system consistency is outstanding.

Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- ☐ Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input
- ■ **Grade 4 - Provides Primary Basis For Application**

2.5 DA – DATA ANALYSIS

The scope of the evaluation for this element of the PSA includes development of component failure rates, maintenance unavailabilities, common cause parameters and miscellaneous data parameters that are needed to calculate the PSA basic event probabilities. The estimation of initiating event frequencies is covered in the initiating events analysis element.

The following items were identified as part of the data review.

- Plant specific data was only collected and analyzed for selected pumps, the diesel generators, and chillers. No justification was given for why plant specific data was not collected for other key components such as valves, breakers, etc. It would be preferable to use the risk importance measures to decide where plant specific data should be collected.

- A rather convoluted procedure was used to apply Bayes' updating to combine plant specific experience on the above mentioned components and generic uncertainty distributions. This procedure was used to take advantage of conjugate properties of certain contributions of assumed prior distributions and assumed likelihood functions. ERIN checked the Bayes' update with a more commonly used procedure where the prior distribution is assumed to be lognormal, the binomial likelihood function is used for demand based failure rates, and the Poisson distribution is used for time based failure rates. The ERIN analysis was performed using a proprietary software known as BART™ (Reference [3]). The results of this comparison are shown in Table 2-5 and show good agreement. CP&L should consider using the recommended procedure as discrepancies in the case of the diesel generator failure rate of approximately 10% could be significant under certain conditions.

Table 2-5

Comparison of Bayes' Updated Failure Rate Distributions

Component/ Failure Mode	Method[1]	Mean	50%Tile	95%Tile	Range Factor
Motor Driven AFW Pump Fail to start (UD-1)	Harris PSA	2.2E-3	1.9E-3	4.5E-3	2.4
	ERIN 1	2.0E-3	1.8E-3	4.1E-3	2.5
Motor Driven AFW Pump Fail to run (UD-2)	Harris PSA	2.6E-5	1.5E-5	8.2E-5	5.4
	ERIN 1	2.6E-5	1.5E-5	8.1E-5	5.3
Turbine Driven AFW Pump Fail to start (UD-3)	Harris PSA	1.2E-2	1.2E-2	1.5E-2	1.3
	ERIN 1	1.4E-2	1.3E-2	2.0E-2	2.1
Containment Spray Pump Fail to run (UD-16)	Harris PSA	2.7E-5	1.6E-5	8.5E-5	5.4
	ERIN 1	2.7E-5	1.6E-5	8.5E-5	5.4
Diesel Generator Fail to Start (UD-17)	Harris PSA	7.1E-3	6.4E-3	1.3E-2	2.0
	ERIN 1	7.8E-3	7.3E-3	1.3E-2	1.9
Chiller Fails to run (UD-20)	Harris PSA	7.8E-5	7.6E-5	1.1E-4	1.5
	ERIN 1	7.8E-5	7.5E-5	1.1E-4	1.5
	ERIN 2	1.0E-4	9.9E-5	1.7E-4	1.8

[1]METHODS: Harris PSA: As calculated in Harris PSA using numerous steps converting between several assumed distributions

ERIN 1: Use of lognormal prior distribution, binomial likelihood for demand, and Poisson likelihood for run; otherwise same generic means, range factors and plant specific evidence as in Harris PSA

ERIN 2: Same methodology as ERIN 1 but with increased range factor to correct inconsistency between assumed generic distribution and plant specific evidence.

- In the analysis of the Chiller failure rate for failure to run (UD-20), the plant specific evidence of 9 failures in 76451 hours whose point estimate is 1.18E-4 per hour seems inconsistent with the assumed prior distribution. The probability of observing so many failures in this exposure time is very low under the assumptions of the prior distribution. In Table 2-5 a revised analysis is performed (ERIN 2) in which the prior distribution has been revised to increase the range factor to 5 to be more consistent with the evidence. As seen in the table the results are sensitive to the assumed prior distribution. The Harris PSA procedure should be revised to check the consistency of the prior distributions and the evidence and adjustments like this should be made before accepting the results of the Bayes update.

- Maintenance data treatment used same inconsistent methods as initiating events; purely generic data for most components and purely plant specific data for selected components. A consistent Bayes' update approach could be used as with failure rates.

- Very good example of how common cause events should be modeled and how data should be screened and mapped for plant specific application. The common cause data source is somewhat outdated and should be updated using the INEEL database.

A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-6.

Table 2-6

Evaluation of Data Analysis Element

Attribute	Assessment
Guidance/ Documentation	It is not known whether CP&L has any separate guidance documents but the Data Analysis Documentation is very thorough and hence, serves as useful guidance to support PSA updates.
Plant Specific Component Data	Generic database for component failures is well documented. Plant specific data was only incorporated for selected components including pumps, emergency diesel generators, and chillers. No valid basis for excluding other risk significant components was provided. The Bayes' update methodology employs a sequence of modeling assumptions to take advantage of conjugate properties, but independent verification of these calculations indicates some discrepancies in comparison with a more straightforward Bayes' update procedure. In one case, chiller failure to run, the plant specific evidence was inconsistent with the assumed generic distributions.
System/ Train Unavailabilities	Plant specific maintenance unavailability data was collected for the same limited set of components as was used for failure rate determination. Generic maintenance data was used for other components. As with the initiating event frequencies, inconsistent methods were used to quantify the plant specific and generic maintenance unavailabilities: classical statistics used for analyzing plant specific data, but Bayesian method used for those quantified using generic data.
Common Cause Failure Quantification	The common cause failure analysis including the selection of components for CCF modeling, plant specific screening and mapping of CCF data and justification and documentation of data interpretation assumptions is excellent. The only aspect of the common cause analysis that could be improved significantly is the incorporation of the INEEL common cause database.
Unique Unavailabilities and Modeling Issues	Unique modeling unavailabilities are documented in the system notebooks and throughout the report. Offsite power recovery should be updated to incorporate the latest industry offsite power recovery data.

Attribute	Assessment
Recommended Enhancements	A more straightforward, less convoluted, and consistent Bayes' update procedure should be used for initiating event frequencies, failure rates and maintenance unavailabilities. The use of narrow generic distributions that are inconsistent with the plant specific evidence should be avoided. Plant specific data should be applied for all risk significant components. The excellent treatment of CCF data should be updated to incorporate the INEEL CCF database.
Overall Process Assessment	This element of the Harris PSA, with the exception of the excellent treatment of common cause, was not up to the level of quality of most other aspects of the PSA. While the data values used in the PSA are reasonable, the data handling methods for failure rates and maintenance could be improved. The current data treatment should be adequate for risk ranking and most risk informed applications but should be enhanced for risk informed applications in which the plant specific equipment performance is an issue.
Recommended Element Grade: ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities ■ **Grade 2 - Supports Risk Ranking Applications** ☐ Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input ☐ Grade 4 - Provides Primary Basis For Application	

2.6 HR – HUMAN RELIABILITY ANALYSIS

The scope of this element includes the systems analyses that support the PSA including the identification of system functions addressed in the PSA, system success criteria, system dependencies, potential for causing an initiating event, fault tree models that support the accident sequence models, and probability models to support the accident sequence quantification. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-7.

Table 2-7

Evaluation of Human Reliability Analysis Element

Attribute	Assessment
Guidance	It is not known whether CP&L has any separate guidance documents but the Human Reliability Analysis Documentation provided in Appendix E is very thorough and hence, serves as useful guidance to support PSA updates.
Pre-Initiator Human Actions	A reasonably complete treatment of pre-initiator Human Actions was provided in Annex A. A good basic treatment of these events is provided and is well documented.
Post Initiator Human Actions	Both ASEP and THERP methods were used for post initiator human actions and these are very well documented in Annex B and C. A good basic treatment of these events is provided and is well documented.
Treatment of Dependencies	Human actions were placed directly on the fault trees, which necessitated an evaluation to see if multiple human errors would impact sequence truncation and quantification. The version of the quantification that was described used a relatively high truncation frequency of $1x10^{-8}$ per year only found two cutsets that had been screened out. This procedure should be extended to a lower truncation no greater than $1x10^{-10}$.
Documentation	The documentation of the human reliability analysis in Appendix E Annex's A, B, C, and D was excellent.
Recommended Enhancements	Dependencies among multiple HEPs were performed for the 1995 model. The relatively high truncation value (1E-8/yr) indicated only two cases to be addressed. It may be prudent to reevaluate this multiple HEP assessment in the future. These multiple operator actions are believed to impact principally those sequences contributing to late containment failure sequences or to SGTR.
Overall Process Assessment	The Harris PSA includes a treatment of human reliability analysis that is sufficient to treat risk-informed applications.

Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- ■ **Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input**
- ☐ Grade 4 - Provides Primary Basis For Application

2.7 DE – DEPENDENCIES

The scope of this element includes the treatment of dependent failures between and among systems and initiating events due to functional, physical, and human sources. Treatment of common cause failures and spatial dependencies due to internal hazards such as flooding are also included in this element. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-8.

Table 2-8

Evaluation of Dependency Analysis Elements

Attribute	Assessment
Guidance/Documentation	It is not known whether CP&L has any separate guidance documents but the Dependency Analysis Documentation is very thorough and hence, serves as useful guidance to support PSA updates.
Dependency Matrices	The Harris PSA did not include dependency matrices, however the system dependencies are delineated separately on a system by system basis in the system notebooks. A more holistic consolidation of dependencies in a single place would make it easier for system engineers to validate plant to model fidelity.
Common Cause Failure Treatment	The common cause failure treatment including a comprehensive set of components and failure modes, and the plant specific screening and mapping of CCF data was excellent.
Spatial Dependencies	The internal flooding analysis is excellent including the screening of flooding scenarios, treatment of a range of flood rates, analysis of spatial interactions, and detailed quantification of high-risk scenarios. The documentation of this aspect of the PSA was the best that this reviewer has seen.
HI Dependencies	The treatment of HI dependencies was adequate but could have been improved by extending the examination of truncated sequences down to a truncation frequency of 1×10^{-10}.
Recommended Enhancements	Add dependency matrices, extend common cause data analysis to include the INEEL common cause data base, and extend the HI dependency treatment to 1×10^{-10} per year.
Overall Process Assessment	The current dependency treatment is adequate to support risk informed applications and with the recommended enhancements would provide the primary basis for application.

Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- ☐ Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input
- © **Grade C4 - Provides Primary Basis For Application (Conditions)**

2.8 ST – STRUCTURAL RESPONSE

The scope of this element includes the structural analyses that support key aspects of the PSA. These structural issues include the success criteria for vessel integrity during severe over pressure and over cooling transients, the capability of the containment in response to severe accident challenges, and the modeling of the response of low pressure piping when exposed to high pressure loads during an interfacing systems LOCA evaluation. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-9.

Table 2-9

Evaluations of Structural Analysis Elements

Attribute	Assessment
Guidance	It is not known whetner CP&L has any separate guidance documents but the PSA documentation of the structural analysis is very thorough and hence, serves as useful guidance to support PSA updates.
RPV Capability	FSAR success criteria are used for RV overpressure protection functions and this represents a potential conservatism in the ATWS evaluation, however due to the low frequency contribution of ATWS this does not distort the results unduly. It does not appear that overcooling transients and reactor vessel failures due to pressurized thermal shock have been modeled, however these sequences even when modeled do not normally contribute noticeably.
Containment Capability	A plant specific containment pressure capacity analysis was performed and the dependence on containment temperature was considered. This aspect of the evaluation was technically sound.
Pipe Overpressurization	The treatment of pipe overpressurization in the interfacing system LOCA evaluation was technically sound.
Recommendations for Enhancements	Maintain this quality level as the PSA experiences future updates
Overall Process Assessment	This element of the PSA is capable of supporting any envisioned risk informed application.
Recommended Element Grade: ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities ☐ Grade 2 - Supports Risk Ranking Applications ■ **Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input** ☐ Grade 4 - Provides Primary Basis For Application	

2.9 QU – QUANTIFICATION

The scope of this element includes the parts of the PSA not considered in the previous elements to support quantification of CDF and interpretation of the results. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-10.

Table 2-10

Evaluation of Level 1 Quantification Element

Attribute	Assessment
Guidance/ Documentation	Very good guidance (HNP-F/PSA-0001) on quantification and recovery. This procedure contains a step-by-step process for duplicating the documented results with the controlled model and quantification files. FLag files, mutually exclusive event files and recovery files are explicitly documented. Minor point, but rearange Table 6-1 in decreasing CDF order instead of alphabetically by sequence name (i.e., same order as Section 6.1.1). The summary of results in Section 1 is weak; at least discuss the top few sequences instead of just showing pie charts and saying, "nothing really dominates - see Section 6."
Dominant Accident Sequences	Very good guidance (HNP-F/PSA-0001) on quantification and recovery. This procedure contains a step-by-step process for duplicating the documented results with the controlled model and quantification files. Flag files, mutually exclusive event files and recovery files are explicitly documented. Minor point, but rearrange Table 6-1 in decreasing CDF order instead of alphabetically by sequence name (i.e., same order as Section 6.1.1). The summary of results in Section 1 is weak; at least discus the top few sequences instead of just showing pie charts and saying, "nothing really dominates - see Section 6."
Truncation/Recovery Analysis	Truncation in the most recent calculation file (2000 model) was 4E-9 in contrast with the 1E-8 level in the PSA report. This is 1E-4 less than CDF and is consistent with current industry standards. The documentation cites the largeness of the SHNPP fault trees and the excessively large number of cutsets that would result from lower truncations as justification. The recovery analysis uses the rule-based EPRI R&R QRECOVER software to apply recoveries to sequences on a detailed, systematic and reproducible basis.
Uncertainty	No statistical uncertainty analysis of the results was apparent.
Results Summary	Documentation in Section 6.1.1 is strong on what has been done. The documentation does not generally cover the underlying reasons or assumptions regarding why certin choices are made. For example, on top sequence TQUB (30.92% of CDF), why is it that fails seal cooling in the internal flooding sequences? Why does the TDEFW pump fail when the battery depletes? The presentation of key assumptions in the the summary is good but a discussion of the impact of these assumptions on the results is missing.

Attribute	Assessment
Recommended Enhancements	Document an uncertainty and sensitivity analysis and provide engineering insights into the results; update the results summary to account for the more recent update.
Overall Process Assessment	The quantification element subject to the recommended enhancements is capable of supporting risk informed applications.
Recommended Element Grade: ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities ☐ Grade 2 - Supports Risk Ranking Applications © **Grade C3 - Supports Risk Significance Evaluations w/Deterministic Input(Conditions)** ☐ Grade 4 - Provides Primary Basis For Application	

2.10 L2 CONTAINMENT PERFORMANCE

The scope of this element includes the interface with the Level 1, the containment event tree structure, the phenomena considered, applicable systems and operator actions, and end states. A summary evaluation of the technical quality and adequacy of this element is provided in Table 2-11. The key points from this evaluation are summarized below. Comments relevant to the overall Level 2 analysis are presented in Section 2.10.1, while detailed comments on the interfacing systems LOCA analysis are provided in Section 2.10.2.

2.10.1 Comments on Overall Level 2 Analysis

The overall Level 2 Analysis is state of the art and is considered a key strength of the Harris PSA. Specific comments are found in Table 2-11.

2.10.2 Comments on ISLOCA Analysis

Due to the importance of ISLOCA sequences as a contributor to LERF for the Harris PSA, a special review was performed to identify the need for any updates to provide a more realistic evaluation of these events.

General Comments in ISLOCA

- There are combinations of failure modes included that do not make physical sense. Until the inboard valve is failed (by rupture, leak, or mispositioning, if possible) the outboard valve(s) have no demand and cannot be questioned. The only way an outboard valve can be "holding" is if the inboard valve has already failed in some way.

- Common cause failures of valves in the rupture failure mode should not be modeled. Any conceivable maintenance errors that might be called common cause rupture would be discovered long before reaching full power operation.

- The rupture and leak failure modes can be combined, since a rupture is just a large sized leak and they are both time-dependent failures. Frequencies can be determined as done for Seabrook, using a Frequency vs. Leak Rate plot developed from actual data.

- The discussion of low pressure pipe failure due to overpressurization should include at least some discussion of other possible pressure boundary failures, i.e., pump seals, bolted flange connections, valve packing, valve body rupture, etc.

RHR Suction Lines

- The model for these lines does not include the leak failure mode. Combining the leak and rupture failure modes would take care of this. If leaks were modeled, they would have to exceed the flow capacity (900 gpm) of the relief valve in the low-pressure portion of the line in order to pressurize the low-pressure piping.

- Rupture of the outboard valve followed by rupture of the inboard valve is not a credible scenario. (See General Comments Above)

- Failure of the inboard valve to hold on demand is not a credible failure mode. The inboard valve always sees the same pressure during power operation.

LHSI Cold Leg Injection Lines

- The failure modes 'Fails Stuck Open' and 'Fails to Hold on Demand' are just different ways to describe 'Fails to Reseat' and should not both be included for these check valves.
- Common cause failures by the rupture failure mode and the leak failure mode should not be included. (See General Comments Above)

Table 2-11

Evaluation of Level 2 Element

Attribute	Assessment
Guidance/ Documentation	It is not known whether CP&L has any separate guidance documents but the PSA documentation of the L2 containment analysis is very thorough and hence, serves as useful guidance to support PSA updates. The documentation is excellent.
Level 1/Level 2 Interface	The interface is provided by a bridge event tree that links the Level 1 ET and Containment Event tree. The Bridge Tree supports the definition and assignment of a comprehensive set of plant damage states that are sufficient to capture severe accident issues relevant to PWRs with large dry containments. This part of the Level 2 analysis is very clearly documented and is technically sound.
Containment Event Tree Phenomena, Systems, Human Actions, Success Criteria	The CET considers all the severe accident phenomena that are expected for this plant and containment type and account for all the relevant NUREG-1150 issues. In addition, direct corium attack of the liner was identified and modeled as a result of plant specific evaluation of containment features. Success criteria for in-vessel recovery, arrest of corium attack of basemat via debris bed cooling, and time windows for restoring vessel and core cooling are reasonable. This part of the Level 2 analysis is very clearly documented and is technically sound.
Containment Capability Assessment	A plant specific probabilistic evaluation of the containment failure modes was performed and used to convolute against the assessed pressure and temperature loads to calculate the containment failure probability. This part of the Level 2 analysis is very clearly documented and is technically sound.
CET End States	The CET release categories provide an adequate spectrum of possible containment releases to support source term definition and calculations.
LERF Definition	The PSA does not calculate a LERF but provides sufficient information on the definition of the release categories to estimate LERF.

Attribute	Assessment
Recommended Enhancements	The interfacing system LOCA frequencies are considered to be conservative due to inclusion of inappropriate common cause failure modes and due to application of conservative data for valve failures. It is expected that an updated evaluation would support up to a one order of magnitude reduction in V-LOCA frequency. Some of the CET probabilities are simply assigned based on qualititatve judgements and could be questioned as to their basis, but seem to be reasonable. More could be done to compare these assessments to industry or NUREG-1150 results, however no significant changes in the results would be expected, only deeper insights into the contributors to the Level 2 results.
Overall Process Assessment	This Level 2 analysis is state of the art for plants in its peer group and is already sufficient to support risk significant determinations with deterministic input. Incorporation of enhancements would bring this PSA element to a level sufficent to provide the primary basis for decision making.

Recommended Element Grade:

- ☐ Grade 1 - Supports Assessment of Plant Vulnerabilities
- ☐ Grade 2 - Supports Risk Ranking Applications
- ☐ Grade 3 - Supports Risk Significance Evaluations w/Deterministic Input
- © **Grade C4 - Provides Primary Basis For Application (Conditions)**

2.11 MU – MAINTENANCE AND UPDATE PROCESS

As part of a normal WOG PSA Peer Review, there is an evaluation of the maintenance and update process. This more complete evaluation involves presentations by the PSA team, discussions regarding applications, review of onsite procedures, etc. Due to the nature of what is involved in this type of review, a peer review was not performed for this element. One comment that is made is that the PSA documentation, particularly the details provided on the use of the CAFTA and R&R Workstation tools to support the various steps of the quantification process, provides excellent guidance for future PSA updates. However, to review this element requires access to information not available to support this peer review, which is based on the review of the existing PSA documentation and models.

SECTION 3
SUMMARY OF RESULTS AND CONCLUSIONS

The results of the evaluation of the Harris PSA elements are summarized in Table 3-1. This table includes the projected PSA Peer Review Grades that the reviewers believe would occur from a full peer review according to the WOG PSA Peer Review Program (Reference [1]). Also included are the strengths and weaknesses of each PSA element that the formal peer review would normally be documented in the Fact and Observation forms. A more complete delineation of these points is found in Section 2.

On balance this PSA is viewed as one of the best-documented PSAs that the reviewers have seen. The systems analysis, thermal hydraulics analysis, containment performance analysis and the dependency analysis were especially well done and are projected for evaluations at grade Level 4 or close to this grade. With the exception of data analysis, which was assessed at grade Level 2, the remaining elements of the PSA were at or near grade Level 3 with only small numbers of issues to clear up in order to achieve this grade level.

The Harris PSA is viewed as capable of supporting risk-informed applications such as the spent fuel pool PSA evaluation that this review was performed in support of. In each application, the applicability of the strengths and weaknesses identified in this review should be reviewed and addressed to determine whether they impact the conclusions of the application. When such impacts are identified, they should be addressed via PSA updates, sensitivity analyses, and/or supplemental engineering analyses as appropriate to support the decisions or conclusions associated with the application. In the opinion of the reviewers, this PSA is in the upper quartile of PSAs in the nuclear industry today; when ranked in terms of the capability to support risk informed applications.

An important finding of the peer review is that the PSA can be used to assess the CDF, Containment Failure Frequency, and Containment Bypass Frequency. If all the specific technical issues raised in this review were resolved, and incorporated into a PSA update, it is expected that the estimated CDF values would be comparable to or lower than those reported in the Harris PSA report, however the uncertainties are larger than those quoted in the report due to the issues noted for the data element. If the issues impacting LERF were addressed in a similar fashion, it is expected that the current LERF results that are supported by the existing PSA be determined to be conservative primarily from conservatisms in the estimation of interfacing systems LOCA frequency.

Table 3-1 Summary of Harris PSA Review Findings

Element Code	PSA Element	Element Grade*	Strengths	Potential Enhancements
IE	Initiating Events	3	Excellent treatment of support system initiating events; Clear functional grouping and binning for sequence model; good use of fault trees for selected initiators	Mixture of classical statistics for some events and Bayes' treatment of other events; problems with system fault trees for initiators
AS	Accident Sequence Evaluation	C3	Very clear documentation of event trees, success criteria and interface with fault tree quantification; good use of ESDs; excellent Level 1/Level 2 interface	Loss of offsite power event tree shifts important details of the logic into the linked fault trees that are more difficult to review.
TH	Thermal Hydraulic Analysis	4	Excellent and traceable documentation	None
SY	System Analysis	4	Excellent and traceable system notebooks	None
DA	Data Analysis	2	Good generic database that is traceable to sources; excellent CCF data treatment	Only limited amount of plant specific data; inconsistent use of Bayes and statistical methods
HR	Human Reliability Analysis	3	Clear and transparent documentation	Treatment of dependencies limited to cutsets > $1x10^{-8}$; need to update offsite power recovery.
DE	Dependencies	C4	Excellent CCF treatment and internal flooding analysis	CCF data source could be updated to latest INEEL CCF; addition of dependency matrices and review by plant staff would ensure PSA reflects as-built plant.
ST	Structural Response	3	Good documentation	None
QU	Quantification	C3	Very good description of quantification process	Results summary includes basic information but is weak on insights; results summary should be updated
L2	Containment Performance	C4	Excellent state of the art treatment of all severe accident phenomena relevant to PWRs with large dry containments; clean Level 1/Level 2 interface.	No credit for SAMGs; conservative treatment of interfacing systems LOCAs; so that results for LERF are conservative
MU	Maintenance Update	N/A	This element was not evaluated in this review	This element was not evaluated in this review

*C indicates the grade is conditional on resolving specific issues noted in the evaluation summaries in Section 2.

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SECTION 4
REFERENCES

[1] Nuclear Energy Institute, "Probabilistic Risk Assessment Peer Review Process Guidance", NEI-002, Draft, 2000.

[2] American Society of Mechanical Engineers, "A Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications(Draft)", Revision 12, May 30, 2000.

[3] INEEL, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants 1980-1996", NUREG/CR-5496, November 1998.

Attachment C

Summary of Reviews to Shearon Harris Nuclear Plant PSA, IPE and IPEEE

Date of Review	Subject and Scope of Review	Organization Performing Review
Nov-00	Peer Review of Shearon Harris PSA The Independent Peer Review team concluded that the Shearon Harris PRA was one of the best-documented PRAs that the reviewers had seen. The systems analysis, thermal hydraulics analysis, containment performance analysis and the dependency analysis were especially well done and were evaluated as grade Level 4 or close to this grade. Grade Level 4 is acceptable for use as a primary basis for developing licensing positions that may change hardware, procedures, requirements, or methods. The data analysis was assessed at grade Level 2. Grade 2 corresponds to the attributes needed for risk ranking of systems, structures, and components. The remaining elements of the PRA were at or near grade Level 3. Grade 3 means that the PRA is adequate to support regulatory applications, when combined with deterministic insights. The ISLOCA analysis was considered quite conservative compared with other similar plants. The ISLOCA initiating event frequency could be reassessed to make the PRA more realistic.	ERIN Engineering and Research, Inc.
Jan-00	NRC Staff's Evaluation of the Shearon Harris Nuclear Plant, Unit 1, Individual Plant Examination of External Events (IPEEE) Submittal (TAC No. M83627) The NRC staff performed a screening review for completeness and reasonableness considering the design and operation of the plant. A Senior Review Board provided further review.	NRC
Apr-98	Shearon Harris Nuclear Plant Probabilistic Safety Assessment Review of Sequence Solutions, Report RSC 98-06, Revision 0 The high-level review of the Harris PSA accident sequence results (CAFTA-generated cut sets) from the model of record assessed the plant system design and the PSA event trees to determine at a qualitative level what results could be reasonably expected. The review also identified the most important accident sequence contributors and determined their applicability based on expected plant response and general PSA modeling guidance.	Ricky Summitt Consulting

Jan-96	NRC Staff's Evaluation of the Shearon Harris Nuclear Plant Individual Plant Examination (IPE Submittal) (Serial HNP-93-835) (TAC No. M74418) The evaluation package consisted of the Staff Evaluation Report (SER), the contractor's Technical Evaluation Reports (TERs) and a summary of the IPE submittal on Internal Events.	NRC
Oct-93	Appendix K, Review Comment Resolution: 1. Initiating Events and Event Sequence Development 2. System Modeling 3. Component Failure Data 4. Human Reliability Analysis 5. Sequence Quantification 6. Documentation 7. System Models 8. Comments from INPO Team member 9. Cutset Review Meeting Appendix K is part of the supporting documentation prepared for the IPE submittal. The review team included CP&L members as well as members from the organizations listed.	1. SAROS 2. SAROS 3. CP&L 4. CP&L 5. CP&L 6. CP&L 7. CP&L 8. INPO 9. SAIC, SAROS, NUS, INPO, CP&L

Attachment D

Plant-Specific Information Provided to ERIN

	Item Description
1.	ASLB Memorandum and Order (Ruling on Late-Filed Environmental Contentions), August 7, 2000, LBP-00-19.
2.	HNP PSA Model of Record (MOR2000) (.CAF, .BE and other related files and calculation HNP-F/PSA-0001, Rev1 documenting the model)
3.	Shearon Harris Nuclear Plant Probabilistic Safety Assessment Fuel Pool Cooling and Cleanup System Notebook, Rev0, RSC 99-14, March 1999 and associated .CAF and .BE files
4.	Fuel Pooling Cooling and Cleanup System Description (SD-116) and Design Basis (DB-110)
5.	Fuel Handling Building Drawings: CAR-2165-G002, G011, G012, G014-G026, G910-G914, G916-G918, F25006, F-25007, F-002506 and F-02507
6.	Heat Load Calculations: ESRs 96-00126, 97-00636, 00-00046
7.	"Containment Overpressure Capacity for the Shearon Harris Nuclear Power Plant, Unit No. 1" by EQE International, march 1993.
8.	HNP Procedures AOP-13, AOP-20, AOP-31, AOP-107 and AOP-116
9.	Reactor Auxiliary Building (RAB) heatup calculation (RAB.ZIP) and Appendix J of the HNP IPE, Rev0, October 1993, ISLOCA analysis from HNP IPE
10.	MAAP input for HNP
11.	Big Rock Point Nuclear Plant Zircaloy Oxidation Analysis prepared for Consumers Power by Sargent & Lundy, SL-5203, April 24,1998.
12.	Shearon Harris Nuclear Plant Probabilistic Safety Assessment Groundrules and Assumptions, Rev0, RSC 97-22, October 1997
13.	MAAP Parameter deck
14.	NUREG/CR-4982 (Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82), July 1987
15.	NUREG/CR-5176 (Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants), January 1989
16.	NUREG/CR-5281 (Value/Impact Analyses of Accident Prevention and Mitigative Options for Spent Fuel Pools), March 1989
17.	NUREG-1353 (Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"), April 1989
18.	NUREG/CR-6451 (A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants), August 1997
19.	NUREG/CR-0649 (Spent Fuel Heatup Following Loss of Water During Storage), March 1979
20.	"Final Draft Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants and Federal Register Notice Requesting Public Comments on Technical Study", February 15, 2000.
21.	NUREG-0575 Volumes 1 and 2 (Handling and Storage of Spent Light Water Power Reactor Fuel), August 1979
22.	MSLB Analysis Inside/Outside Containment (Raytheon =Washington Nuclear), Report No. 78704.1.3.2.01.10, August 13, 2000.
23.	NUREG-0575 Volume 3, August 1979.
24.	MAAP input deck

25.	RMA-028, "Review of WOG Severe Accident Management Guidelines as Applied to Shearon Harris", Rev0, July 1996.
26.	RMA-032, "Development of Shearon Harris Severe Accident Calculational Aids", Rev0, July 1996.
27.	Susquehanna Steam Electric Station, Units 1 and 2, Draft Safety Evaluation Regarding Spent Fuel Pool Cooling Issues (TAC. No. M85337), October 25, 1994.
28.	SECY-99-168, Improving Decommissioning Regulations for Nuclear Power Plants, December 21, 1999.
29.	AOP-013, Rev11, Fuel Handling Accident
30.	OP-112, Rev13, Containment Spray System
31.	OP-116, Rev17, Fuel Pool Cooling and Cleanup
32.	OP-143.3, Rev19, Demineralized Water
33.	HNP spent fuel pool drawings: CPL-2165-S-0805Rev7 and CPL-2165-S-0807Rev4.
34.	HNP IPE Appendix J, Attachment 3 (hardcopy of material not in APP_J.ZIP)
35.	HNPsum2000.xls, PSA summary of release categories
36.	NRC letter to David Lochbaum, Spent Fuel Pool Cooling Generic Review (TAC No. M88094) including 2 attachments: 1. AEOD/S96-02, Assessment of Spent Fuel Cooling, September 1996. 2. INEL-96/0334, Loss of Spent Fuel Pool Cooling PRA: Model and Results, September 1996. 3. The Potential for Propagation of the Self Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Storage Pool, Pisano, et al., Draft - Jan 1984
37.	SF-0040, Rev0, Spent Fuel Pools C and D Activation Project Thermal Hydraulic Analysis, November 10, 1998.
38.	NRC Report 7590-01-P, Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants (Draft for Comment), February 2000.
39.	Shearon Harris Nuclear Power Plant, Unit 1, Individual Plant Examination for External Events (IPEEE) Submittal, June 1995.
40.	NRC's Staff Evaluation of the Shearon Harris Nuclear Power Plant, Unit 1, Individual Plant Examination for External Events (IPEEE) Submittal (TAC No. M83627), January 14, 2000.
41.	IPE and supporting documentation
42.	Drawings: CAR-2165-G-011 through 021; CAR-2165-G-151 through G-156 and CPL-2165-S1308, G-808, S-1324, -G-824, S-1300, -G-800
43.	HNP Periodic System Review, Spent Fuel Pool Cooling (7110), June 20, 2000.
44.	HNP AMMS printout for System 7110 (Spent Fuel Pool Cooling) for 1993 through 1998.
45.	Various plant drawings – see attached table for complete listing.
46.	HNP MAAP runs/card image input decks
47.	HNP Procedure OMP-003, Rev12, Outage Shutdown Risk Management
48.	HNP Procedure OMP-004, Rev8, Control of Plant Activities During Reduced Inventory Conditions
49.	HNP Technical Specification 3/4.9.4 and Bases
50.	Assumed duration that containment is not isolated during refueling outage.
51.	Response to ERIN request: JPMs; Demin Flow; TSC guidance; Containment isolation assumption; HVAC characteristics
52.	Response to request for additional information
53.	Fire cutsets Non-fire cutsets
54.	Basis for HNP fuel pool cooling pump unavailability. Basis for skimmer and purification pump values.

55.	HNP Steam Generator and IPEEE assumptions
56.	Power Source Locations
57.	IN 2000-13, NRC Information Notice, "Review of Refueling Outage Risk", September 27, 2000.
58.	Dose Calculation
59.	Source Term (not provided to ERIN; however, implicit in the dose calculation)
60.	Clarification: IPEEE/Tier 2 information Fire Area 1-A-4- COMB
61.	Availability of Fire Pumper Truck following seismic event
62.	Assumptions for RAB adverse environment
63.	System Description: SD-145, Rev5, Component Cooling Water System
64.	System Description: SD-156, Rev8, Plant Electrical Distribution System
65.	Operating Procedure: OP-145, Rev26, Component Cooling Water
66.	System Description: SD-139, Rev12, Service Water System
67.	CCW pump motors and pumps, EQ classification
68.	Power supply for Normal Service Water pumps
69.	Power Supply Question – FHB Crane
70.	LER 89-002-00 Spent Fuel Pool Draining
71.	Control Room Habitability – NEI Summary
72.	Extract from October 6, 2000 ACRS-Commissioners meeting
73.	FSAR 2.3.2 and wind rose data
74.	EOP-PP-013, Rev5, LOCA Outside Containment
75.	EOP-PP-012, Rev12, Loss of Emergency Coolant Recirculation
76.	Fire Brigade entry into FHB in a 190F environment
77.	PLP-201, Rev39, Emergency Plan
78.	Use of SFP as makeup source for core not identified in response guidance
79.	Dose calculations for 4 cases: Early and Late Containment Failure, Containment Isolation Failure and ISLOCA
80.	Reference for "Human Tolerance for Heat"
81.	Location of Water Treatment Building
82.	SGTR dose cases
83.	Revised Time-to-Boil Calculations
84.	Procedure FPP-013, Rev28, Fire Protection – Minimum Requirements and Mitigating Actions and recent history of Holly Springs FD backup
85.	Location of Water Treatment Building and other site structures
86.	Revised access times
87.	Revised access times – Chi/Q based on 10-year data summary

Drawings provided to ERIN [redacted]
CAR-2168-G-500
CAR-2168-G-501
CAR-2168-G-501S02
CAR-2168-G-501S03
CAR-2168-G-501S04
CAR-2168-G-501S07
CAR-2168-G-501S09
CAR-2168-G-501S12
CAR-2168-G-501S13
CAR-2168-G-501S14
CAR-2168-G-501S15
CAR-2168-G-502S01
CAR-2168-G-502S02
CAR-2168-G-502S03
CAR-2168-G-502S04
CAR-2168-G-503S01
CAR-2168-G-524S03
CAR-2168-G-524S04
CAR-2168-G-532S05
CAR-2168-G-533
CAR-2168-G-539S03
CAR-2168-G-540S01
CAR-2168-G-555S06
CAR-2168-G-612S01
CAR-2168-G-800
CPL – 2165-S-2307
CPL – 2165-S-2308
CPL – 2165-S-2309
CPL – 2165-S-2310
CPL – 2165-S-2311
CPL – 2165-S-2312
CPL – 2165-S-2313
CPL – 2165-S-2314
CPL – 2165-S-2315

EXHIBIT 4

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

AFFIDAVIT OF R. STEVEN EDWARDS

COUNTY OF WAKE	)	
	)	ss:
STATE OF NORTH CAROLINA	)	

I, Robert Steven Edwards, being sworn, do on oath depose and say:

1. I am a resident of the State of North Carolina. I am employed by Carolina Power & Light Company ("CP&L") and work at the Harris Nuclear Plant ("HNP" or "Harris Plant" or "Harris") in the Nuclear Engineering Department. Presently, I am the Supervisor, Spent Fuel Pool Project, and am responsible for commissioning and placing into service Harris spent fuel pools C and D. My business address is 5413 Shearon Harris Road, New Hill, North Carolina 27562-0165.

2. I was graduated from North Carolina State University in 1982 with a B.S. in Industrial Engineering. Since graduation, I have been employed by CP&L, first

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as an Associate Engineer, then Engineer, at the Robinson Nuclear Plant, responsible for planning, scheduling and execution of outages and major projects. Beginning in 1986, I served in the Technical Support Unit at Robinson as a System Engineer – Mechanical Systems. Promoted to Senior Engineer in July 1988, I supervised a staff of contract engineers responsible for specific projects at Robinson. In June 1991, I assumed the position of Project Engineer – Mechanical Systems at Robinson and managed a staff of four system engineers and two component engineers responsible for the operation, performance, reliability and maintenance of various plant systems. In August 1992, I became the Director – Information Architecture (Nuclear) in CP&L's Corporate Management Services and served as the management-level liaison and project manager for nuclear-related information technology projects at CP&L's nuclear plants. In October 1994, I moved to the position of Director – Project Control in the Corporate Nuclear Business Operations Group. In that position, I facilitated the development of long-range planning at each CP&L nuclear plant and provided oversight and administration of project management and economic evaluation processes and activities. In July 1996, I moved to Corporate Nuclear Engineering and became Manager of Projects, responsible for scope, cost, schedule, and quality of various nuclear projects. In April 1998, I was assigned to the Harris Major Projects Section and became responsible for the spent fuel pool C and D activation projects, including the completion of the spent fuel pool cooling and cleanup system ("SFPCCS"), spent fuel storage rack design and installation, and related activities. My resume is provided as Attachment A to this affidavit.

3. The purpose of this affidavit is to set forth the data and calculations on which CP&L relies in establishing the time to heat up the Harris spent fuel pools to boiling, and after boiling has started, the additional time necessary to then boil the coolant level down to the top of the spent fuel racks. First, I summarize the background of the license amendment request and the information submitted in support of the application. Second, I describe the Harris spent fuel pool physical arrangement and associated equipment. Third, I discuss the types of heatup calculations used and their applicability to the Harris spent fuel pools. Fourth, I discuss the data and assumptions used in calculations. Finally, I describe the results of the calculations.

BACKGROUND

4. CP&L's application for a license amendment to place spent fuel pools C and D in service was submitted on December 23, 1998. As the project manager for the Harris spent fuel pool C and D activation projects, I was responsible for development of the factual information set forth in the license amendment request. The information in the license amendment request, as updated by additional information subsequently submitted, is accurate to the best of my knowledge and belief.

5. The license amendment request and the need to expand spent fuel storage at Harris results from the failure of the U.S. Department of Energy ("DOE") to begin taking delivery of spent fuel in 1998, as required by the contract between DOE and CP&L and by the Nuclear Waste Policy Act of 1982, as amended. CP&L originally requested that the license amendment to allow placement of spent fuel in spent fuel pools C and D be issued no later than December 31, 1999, as CP&L had planned to begin loading spent fuel in pool C starting in 2000. Further delays threaten to adversely impact CP&L's ability to maintain adequate spent fuel storage capacity and, with the loss of full core discharge capability at one or more of CP&L's nuclear plants, could lead to a forced shutdown condition.

6. The NRC Staff reviewers requested additional information regarding the license amendment request by letters dated March 24, 1999, April 29, 1999, June 16, 1999, August 5, 1999, September 20, 1999, and by conference calls on March 30, 2000 and April 4, 2000. CP&L responded to each request for additional information ("RAI") respectively on April 30, 1999, June 14, 1999, July 23, 1999, September 3, 1999, and October 29, 1999, and April 14, 2000. CP&L also provided additional information to the NRC Staff on October 15, 1999 and July 19, 2000, to supplement previous responses. As the project manager for the Harris spent fuel pool C and D activation projects, I was responsible for development of the factual information set forth in the responses to the NRC

Staff. Following the Board's Memorandum and Order dated August 7, 2000, I was also responsible for development of factual information responsive to the NRC and BCOC discovery requests. The information in each of those responses, as supplemented, remains accurate to the best of my knowledge and belief.

7. I have previously given an affidavit in this matter on December 30, 1999. The information in that affidavit remains accurate to the best of my knowledge and belief.

8. Harris was originally planned as a four nuclear unit site (Harris 1, 2, 3 and 4). Harris 3 and 4 were canceled in late 1981. Harris 2 was canceled in late 1983. Spent fuel pools A, B, C and D and the spent fuel pool cooling and cleanup system ("SFPCCS") for spent fuel pools A and B were completed as part of the Harris fuel handling building, are described in the Harris Final Safety Analysis Report ("FSAR"), and are licensed as part of Harris.

9. Construction on the SFPCCS for spent fuel pools C and D was discontinued after Harris 2 was canceled. By that time, concrete had been poured, all four spent fuel pools had been constructed, and the SFPCCS piping immediately outside and under the spent fuel pools was installed, welded in place and embedded in reinforced concrete. The SFPCCS for spent fuel pools A and B was completed and placed in service. Harris 1 began commercial operations in 1987. Sometime in late 1988 or 1989, before the first discharge of spent fuel and refueling of Harris 1, spent fuel pool A was filled with borated water. Spent fuel pool B was

filled with borated water on or about the time spent fuel was first discharged from the Harris reactor. Because spent fuel pools C and D are connected to spent fuel pools A and B by transfer canals, at some point in or after 1989, spent fuel pools C and D were also filled with borated water to allow the gates in the transfer canal to be opened without a loss of water and preclude an inadvertent partial drain-down of spent fuel pools A and B to spent fuel pools C and D.

HARRIS PLANT SPENT FUEL POOLS AND ASSOCIATED SYSTEMS AND EQUIPMENT

10. As the project manager for the activation of spent fuel pools C and D, my work encompasses analytical design and engineering evaluations, management of the hands-on physical implementation of the modifications to the SFPCCS, and inspection and preparation of the spent fuel pools themselves. As a consequence of my extensive work at Harris and with the Harris spent fuel pools, I am familiar with the physical layout, system configurations, equipment installations, operations, and operating procedures for Harris, as they relate to normal and alternate operation of the fuel handling building, the spent fuel pools, and associated support systems and equipment.

11. As Harris was originally envisioned as a four unit facility with a shared fuel handling building, the fuel handling building was designed and constructed with four separate pools capable of storing spent nuclear fuel. Spent fuel pools A and B were originally intended to support Harris Units 1 and 4. Spent fuel pools C

and D were originally intended to support Harris Units 2 and 3. In addition, the fuel handling building contains a cask unloading pool, which can be connected to any spent fuel pool through transfer canals.

12. The layout of the Harris fuel handling building is illustrated and described in detail in the Harris FSAR, sections 3.8.4.1.3 and 9.1. Each spent fuel pool and the cask unloading pool are interconnected by a main transfer canal, oriented in a north-south direction, and two fuel transfer canals, oriented east-west. The spent fuel pools and transfer canals contain sufficient amounts of water to facilitate safe fuel handling and storage activities. The spent fuel pools, transfer canals, and cask unloading pool contain openings for the underwater movement of fuel assemblies between the pools and transfer canals. These openings also allow the communication of water between the pools and transfer canals. Removable bulkhead gates are installed in the openings when there is a need to isolate a particular pool or canal from the others. The isolation function of the bulkhead gates is provided by stainless steel structural components and inflatable seals, which are installed around the sides of the gates that fit into slots in the pool and canal openings. The seals are normally inflated using instrument air supplied at the installed gate location.

13. The normal configuration of the spent fuel pools (*i.e.*, the configuration present 99% of the time an on annual basis) is with open communication (*i.e.*, the gates removed) between spent fuel pools A and B and the interconnecting south ("Unit

1/4") transfer canal. Plant Operating Procedure OP-116, section 8.27 requires that a "clear path" be maintained between pools A and B. The cask unloading pool is normally open with the gate removed between it and the north transfer canal. Spent fuel pools C and D are currently normally isolated from the main transfer canal and spent fuel pools A and B. This alignment is illustrated in Attachment B. The expected configuration of the spent fuel pools (*i.e.*, the configuration expected to be present 99% of the time on an annual basis) following approval of the pending license amendment request is with open communication between spent fuel pools A and B, the connecting transfer canal and the main transfer canal. The cask unloading pool will normally be connected to spent fuel pools C and D through their interconnecting north ("Unit 2/3") fuel transfer canal. Spent fuel pools C and D, and the cask unloading pool will be isolated from pools A and B by a gate installed at the cask unloading pool end of the main fuel transfer canal. This alignment is illustrated in Attachment C.

14. The original Harris design included a SFPCCS to service spent fuel pools A and B, and a separate SFPCCS to service spent fuel pools C and D. The SFPCCS for spent fuel pools A and B is in service. The SFPCCS for spent fuel pools C and D was not completed, but will be finished and placed in service to support spent fuel operations pursuant to the pending license amendment request.

SPENT FUEL POOL HEATUP CALCULATIONS

15. I directed that calculations be performed to determine two values for the Harris spent fuel pools: (1) the time to heat up the individual pools (A, B, C, D) to boiling temperature (*i.e.*, 212 degrees Fahrenheit) and (2) the additional time to boil the coolant level down to the top of the spent fuel racks. In addition, I directed calculation of the amount of water required to offset the boiling rate in gallons per minute ("gpm") for each case. In turn, I used these calculations to perform a "best estimate" analysis, meaning that assumed input values are based on normally expected operating conditions based on historical data and plant operating records.

16. As a first step, calculations using standard, commonly used heat transfer equations were performed. The individual calculation steps are described in Attachment D. The heatup and boiloff calculations were performed by Andrew Howe, a degreed nuclear engineer and civil engineer in the Harris Engineering Support Section with 18 years experience performing these types of calculations. Mr. Howe has been previously licensed as a Senior Reactor Operator and is currently assigned as the Harris spent fuel pool cooling and cleanup system engineer. The methodology and inputs were independently reviewed by Tom Scattergood, a second qualified engineer in the Harris Engineering Support Section. Mr. Scattergood has Bachelors and Masters degrees in mechanical engineering and over eight years experience performing this type of work.

17. I then used these heatup and boiloff calculations to prepare an analysis responsive to the Board's request for a "best estimate" analysis by revising the performance assumptions to reflect normally expected operating conditions based on historical data and plant operating records. My analyses resulted in three values for each scenario: (1) time to heat the pools to boiling temperature, (2) additional time to boil the water down to the top of the racks, and (3) makeup flowrate required to offset boiling. The calculations I performed were independently reviewed by Edison Morales, a licensed professional engineer and Harris mechanical engineer with over 29 years of relevant experience.

18. The final calculations were performed using a Microsoft Excel spreadsheet. Results were independently verified using manual techniques and a hand calculator.

DATA AND ASSUMPTIONS

19. The input values and initial conditions for the heatup calculations were obtained from several sources. A complete list of calculation input values and sources is provided as Attachment E.

20. Initial condition assumptions were based on current knowledge, existing license and administrative controls, and professional judgments on expected future operating conditions. In each case where a future condition could not be definitively established, a best estimate assumption of that condition was used. Key assumptions in this category include: (a) Beginning of Cycle heat load for

spent fuel pools A and B is assumed to be a base heat load from HNP Calculation SF-40, Spent Fuel Pools C and D Activation Project Thermal Hydraulic Analysis, dated November 10, 1998, which includes a one-third freshly discharged core after startup from a refueling outage; (b) spent fuel pool temperature at the initiation of the heatup is the temperature expected based on Harris historical operating records; (c) spent fuel pools C and D heat load is either 1 MBTU/hr (the maximum allowed by the pending license amendment request) or 15.6 MBTU/hr (the maximum calculated heat load with both pools C and D completely filled); and (d) the water level in the spent fuel pool and transfer canals is the normal pool level expected during plant operation. In addition, several conservative assumptions are incorporated in the heatup calculations, including: (a) water volume in the cask unloading pool was not considered (*i.e.*, gate 8 is assumed installed); (b) no credit is taken for heat transfer to the pool liners, concrete structure, or atmosphere; (c) no credit is taken for any makeup water addition after the initiation of the heatup.

21. Initial operating conditions for the spent fuel pools and transfer canals were established as the lineup expected during normal operation of the plant (*i.e.*, the configuration expected 99% of the time on an annual basis). In all scenarios, spent fuel pools A and B are interconnected through their transfer canal and connected to the main transfer canal (*i.e.*, gate 2 is installed and gates 1, 3 and 4 are removed). For the 1 MBTU/hr scenario involving spent fuel pool C, it is

interconnected with its transfer canal and isolated from the main transfer canal (*i.e.*, gate 7 removed and gates 5 and 6 installed). Spent fuel pool D is not considered (*i.e.*, gate 9 is installed) in the 1 MBTU/hr scenarios, as all the racks and fuel will be in spent fuel pool C. Gate 8 is also assumed to be installed, as the inventory of the cask unloading pool is conservatively not credited in the calculations. In the 15.6 MBTU/hr scenario, racks and fuel will be installed in spent fuel pools C and D and the pools will be interconnected through their transfer canal, but isolated from the main transfer canal (*i.e.*, gates 7 and 9 are removed and 5 and 6 are installed) and the cask unloading pool (*i.e.*, gate 8 installed to reflect the conservative assumption not to credit this water volume).

CALCULATION RESULTS

22. A table listing each heatup calculation and the results for each analyzed scenario is included as Attachment F. These calculations determined that if all cooling and makeup are lost to spent fuel pools A and B at the beginning of an operating cycle, it would take approximately 20.5 hours for the pools to heat up to boiling temperature. Once boiling begins, it would take an additional 7.2 days for the water in the pools to boil down to the top of the racks where uncovery of fuel could begin. During this period, approximately 54 gallons per minute of makeup water would needed to maintain water level in the pools constant. If all cooling and makeup is lost near the end of an operating cycle (when the heat load from the most recently discharged fuel has diminished significantly), it would take

approximately 38 hours for spent fuel pools A and B to heat up to boiling. Once boiling begins during this scenario, it would take an additional 13.5 days for the water in spent fuel pools A and B to boil down to the top of the racks. Approximately 29 gallons per minute of makeup water would need to be added to the pools to maintain water level in the pools constant. In the 1 MBTU/hr scenario (involving only spent fuel pool C), it would take 16 days for water in spent fuel pool C to reach boiling temperature if all cooling and makeup water is lost. Once boiling in spent fuel pool C begins in this scenario, boiling would have to continue uninterrupted for an additional 99 days for the pool C water level to boil down to the top of the racks. During this period only slightly more than two gallons per minute of makeup water would be necessary to offset the boiling rate. For the maximum calculated end of life heat loads in spent fuel pools C and D, the pools will heat up to boiling temperature in approximately 34 hours following loss of all cooling and makeup. Once boiling begins in the pools in this scenario, an additional 8.8 days without cooling or makeup is required for the water in spent fuel pools C and D to boil down to the top of the racks. During this period, approximately 34 gallons per minute of makeup water would needed to maintain water level in the pools constant. These calculations are in Attachment G.

23. The calculations performed to determine the best estimate time to boil, additional time to boil to the top of the racks, and makeup required to offset the boiloff rate

under different heat load scenarios were all performed using standard heat transfer techniques. These techniques are well understood and straightforward. These calculations used the same approach that has been historically employed at the Harris Plant to perform similar calculations. These calculations were performed by experienced and qualified engineers knowledgeable in both heat transfer and Harris plant design and operation. In addition, the inputs, methodologies and results were independently reviewed by experienced and qualified engineers.

24. As an additional check, the results of these calculations were compared for consistency with the spent fuel pool heat up rates identified in FSAR 9.1.3.3. (Amendment No. 50). The information contained in this section of the FSAR discusses expected heat loads and heat up rates through operating cycle 10. The FSAR identifies that under these conditions (which includes an assumed heat load in spent fuel pools A and B of 16.84 MBTU/hr) spent fuel pools A and B would heat up from 112.7°F to 137°F in 5.56 hours, which equates to a heat up rate of 4.37°/hr. The best estimate time to boil calculations determined that spent fuel pools A and B would heat up from 95°F to 212°F in 20.57 hours for the 25 MBTU/hr beginning of cycle heat load scenario, which equates to a heat up rate in spent fuel pools A and B of 5.69°/hr. The spent fuel pool A and B end of cycle scenario calculations determined that spent fuel pools A and B would heat up from 95°F to 212°F in 38.67 hours, which equates to a heat up rate of 3.03°/hr.

Based on these results, the minimum and maximum heat up rates calculated as best estimates range from 3.03°/hr to 5.69°/hr compared to a value based on the FSAR of 4.37°/hr. These values are consistent and the difference, to the extent they are significant at all, appear primarily because of different heat load assumptions (*i.e.*, 25 MBTU/hr and 13.3 MBTU/hr heat loads used in the best estimate calculations assume spent fuel pools A and B are 'full' at the beginning and end of an operating cycle. The 16.84 MBTU/hr heat load identified in the FSAR is a bounding heat load calculated through operating cycle 10, which is the current Harris fuel cycle). This comparison provides me a high level of confidence that the best estimate analyses produced results that appropriately characterize the expected plant performance under the postulated conditions.

25. The results of these calculations show that in the highly unlikely event that all cooling and makeup to the spent fuel pools is lost, a considerable amount of time is available for Harris operators to re-establish cooling or makeup flow in order to prevent the spent fuel pool water level from boiling down to the point where fuel uncovery could occur. Even considering the worst case scenario where cooling and makeup is lost to 'full' spent fuel pools A and B at the beginning of an operating cycle, then the Harris Plant operators would have over 20 hours to take actions necessary to establish makeup or cooling flow before spent fuel pools A and B reached boiling temperatures. The plant operators would then have approximately an additional week to re-establish spent fuel pool cooling or make

up before the water level boiled down to the top of the fuel racks. Under this scenario, the operators would need to provide less than 54 gallon per minute of make up water in order to offset the boiling rate in spent fuel pools A and B. Since heat loads in spent fuel pools C and D are less than the corresponding heat loads in pools A and B, an even longer time is available to establish cooling or makeup flow to these pools.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2000.

R. Steven Edwards

R. Steven Edwards

Subscribed and sworn to before me
this 15 day of November 2000.

Darlene S. Yarborough

My Commission expires: 2-21-2005

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

R. Steven Edwards

Summary: Eighteen years experience in engineering, project management and outage management.

EXPERIENCE: Carolina Power & Light Company, June 1982 - Present

Supervisor, Spent Fuel Pool Project, Harris Plant, Nuclear Engineering (April 1998 - Present)

Project manager for Harris spent fuel pool 'C' and 'D' activation projects including spent fuel pool cooling and cleanup system completion, spent fuel storage rack design and installation, pool cleanup, and related activities. Responsible for all aspects of scope, cost, schedule and quality of projects. Responsible for study, design and implementation activities. Supervise multi-disciplined modification engineering staff that includes mechanical, civil and electrical engineers that develop plant design change modifications, oversee architect/engineer designs, write procedures, perform 10CFR50.59 analyses, perform ANSI N45.2.11 design verification reviews, and perform owner reviews of A/E developed modifications and calculations. Manage activities of various A/E engineers performing design activities including Bechtel, Sargent & Lundy, Duke Engineering, Raytheon, Protopower and Holtec. Responsible for development of License Amendment Request for SFP Activation project. Provide technical support to spent fuel communications team. Perform root cause evaluations. Serve as Emergency Response Organization Company Technical Spokesperson.

Manager of Projects, Nuclear Engineering (July 1996 - April 1998)

Project manager responsible for scope, cost, schedule and quality of various nuclear projects. Responsible for A/E design and analysis. Managed outsource engineering activities (scope development, schedule & cost management, AE negotiations & interface) for preferred and specialty engineering AE's and contractors. Provided group-wide oversight and administration of project management and economic evaluation processes, procedures and activities. Responsible for three-phase project authorization including value-added technical and financial review of projects requiring executive approval. Delivered economic evaluation module at NGG Business Concepts Course. Taught Project Cost Management module for Project Management Institute (PMI) project manager certification course. Developed and delivered various project management/ project controls presentations to industry groups such as Integrated Scheduling & Planning Utility Group (ISPUG) and Institute for International Research Budgeting and Forecasting Conference.

Director - Project Control, Nuclear Business Operations/ Operations & Environmental Support (October 1994 - July 1996)

Provided group-wide oversight and administration of project management and economic evaluation processes and activities. Lead development of NGG project management procedure. Responsible for three-phase project authorization. Developed and delivered project management and economic analysis training to plant personnel focusing on fundamentals and NGG specifics. Delivered various project management related presentations to industry groups and internal company management. Managed implementation of integrated project cost/schedule reporting system that combined FAIM financial data with Prestige schedule information. Developed and delivered economic evaluation module of NGG Business Concepts Course. Managed project budgeting team that implemented process to use Prestige schedule and resource data to build budget for plant projects. Facilitated development of Long Range Planning process at each nuclear plant. Project management peer group facilitator.

R. Steven Edwards

Director - Information Architecture (Nuclear), Management Services (August 1992 - October 1994)

Served as management-level liaison and project manager for nuclear related information technology projects. Provided technical and business process perspective for corporately implemented nuclear I/T projects. Coordinated the development of the nuclear portion of the Corporate Information Technology (I/T) Plan including administration of project prioritization process. Evaluated NGG generated requests for I/T products and services including evaluation of business justification, development of cost/benefit analyses and approval of I/S resource allocations.

Project Engineer - Mechanical Systems, Technical Support, Robinson Plant
(June 1991 - August 1992)

Managed staff of four system engineers and two component engineers responsible for operation, performance, reliability and maintenance of various plant NSSS, support and secondary mechanical systems and equipment such as high head safety injection, low head SI/residual heat removal, containment spray, reactor coolant pumps, liquid & gaseous waste disposal, steam generator blowdown, HVAC, make up water treatment, condensate polishing, etc. Provided extensive coaching and mentoring to staff with varied experience/education levels in development of their customer focused, performance oriented system and component engineering skills. Served as refueling outage Technical Support Shift Manager responsible for timely and successful completion of all engineering related outage activities through coordination of efforts with operations, maintenance, corporate engineering and other site management as well as supervision of engineers assigned to emergent activities and planned projects. Served on Emergency Response Organization as Accident Assessment Team - Mechanical Engineer and Emergency Communicator.

System Engineer - Mechanical Systems, Technical Support, Robinson Plant
Senior Engineer (July 1988 - June 1991); Engineer (November 1986 - July 1988)

Supervised staff of contract engineers responsible for specific projects including plant performance monitoring, procedure rewrite, backlog assessment, engineering training program, and work management system development (1990-1991).

System engineer responsible for operation, performance, reliability and maintenance of various mechanical systems including all plant HVAC, containment vessel (civil and support systems), LHSI/RHR, containment spray, post accident containment venting/H2 recombiner, primary and post-accident sampling, etc. (1986-1990). As system engineer, monitored system/equipment performance; performed surveillance tests; developed engineering evaluations, temporary plant modifications, procedures, 10CFR50.59 safety analyses, ANSI N45.2.11 design verification reviews, procurement engineering reviews, etc. Provided oversight to maintenance staff in troubleshooting system/equipment problems. Conducted root cause analyses. Served on Emergency Response Organization as Accident Assessment Team - Mechanical Engineer and Emergency Communicator.

Outage Planning and Scheduling Engineer, Outage Management, Robinson Plant
Engineer (June 1984 - November 1986); Associate Engineer (June 1982 - June 1984)

Responsible for planning, scheduling and execution of outages and major projects.

R. Steven Edwards

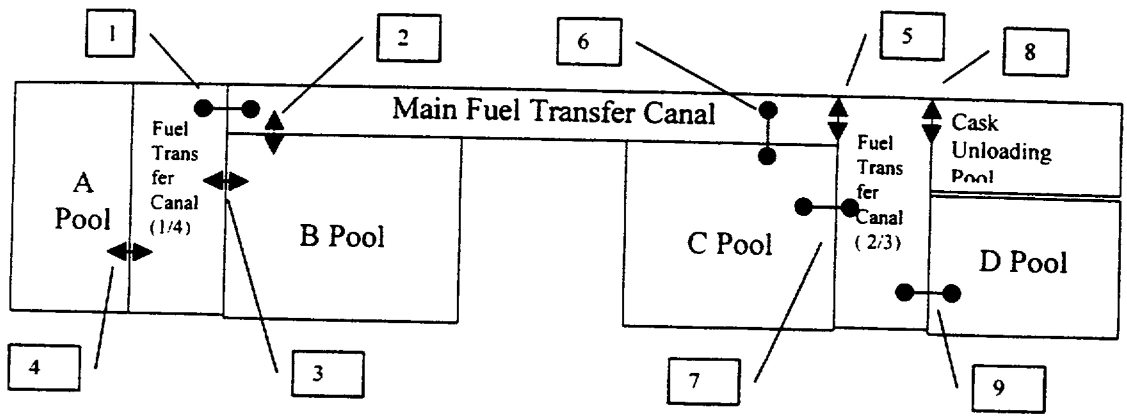
Developed detail and summary level schedules for forced outages, refueling outages, steam generator replacement outage and normal operating periods using manual CPM and ARTEMIS project management system. Led plan-of-day meetings. Served as field coordinator in outage management organization for major projects such as S/G eddy current.

PROFESSIONAL DEVELOPMENT: Attended American Management Association Project Management and Financial Analysis training, Reengineering Fundamentals Seminar, Harvard University In-Place Filter Testing Workshop, industry sponsored ANSI N510 Fan and Filter Testing Workshop, and NCSU Fundamentals of HVAC Design. Participated in company sponsored technical, project management and management/supervisory development training. Engineer in Training Certification - State of North Carolina.

EDUCATION: Bachelor of Science in Industrial Engineering, North Carolina State University, May 1982

Attachment B

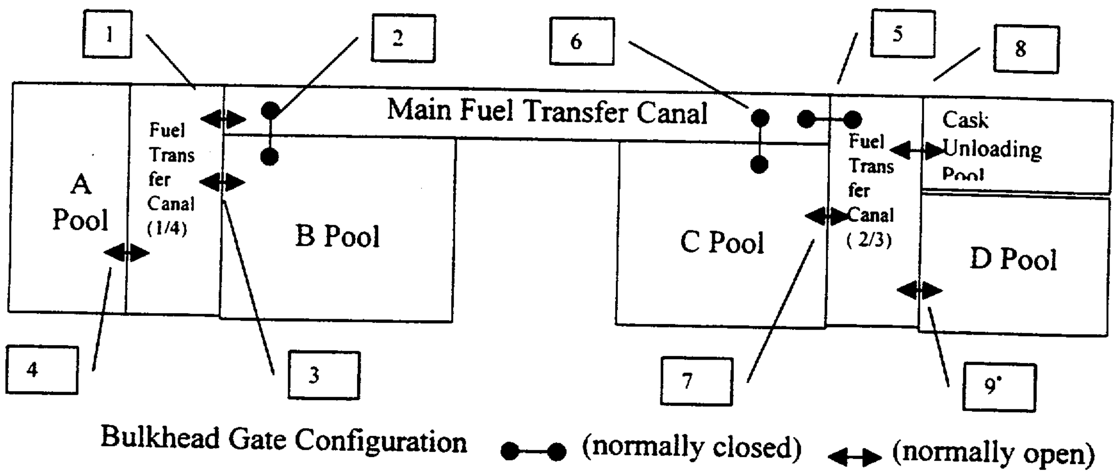
Diagram Illustrating HNP Spent Fuel Storage Pools, Transfer Canals, and Current Bulkhead Gate Configuration.



Bulkhead Gate Configuration (normally closed) (normally open)

Attachment C

Diagram illustrating anticipated bulkhead gate configuration in the HNP spent fuel pools subsequent to operational use of C and D pools.



* The "normally open" configuration for gate 9 would apply subsequent to placing this pool in service that is scheduled for early the next decade. Otherwise, this gate would remain normally closed.

Attachment D

Description of the Key Steps in the Spent Fuel Pool Heatup Calculations

The following steps were taken in the calculation of spent fuel heatup and boiling:

Water volume

The amount of water available in the pools was determined by calculating the volume of the pools and then subtracting out the volume of the racks and the spent fuel. Since pools A and B are operated such that the pools remain interconnected and connected to the main transfer canal during all normal conditions, then the combined volume of the two pools, the main transfer canal and the Unit 1/4 transfer canal was considered together for these calculations. For the spent fuel pool A and B time to boil calculations, the water volume from the main transfer canal was not included; it was assumed that, due to its length and configuration, the temperature in the main transfer canal would lag behind the temperature in the pools during the heatup. However, the volume of water in the main transfer canal was included during the calculation of the additional time required to boil down to the top of the racks, since this water volume would definitely be available to displace water in the pools during the period when the boiling is actually occurring. Once both pools C and D are placed in service these pools will operate such that they will remain interconnected. Therefore, the combined volume of pools C and D and the Unit 2/3 transfer canal was considered together for these calculations.

Heat input required to raise the temperature of this water volume to boiling.

In the highly unlikely event that all cooling and makeup to the pools is lost, the heat input into the water from the spent fuel will cause the water temperature to increase. The amount of heat input required to increase the temperature of this volume of water from the normal expected temperature to boiling temperature was calculated using standard heat transfer equations.

Time required to reach boiling.

The heat input required to reach boiling divided by the heat load present in the pools yields the time required to reach boiling temperature. Best estimate heat loads were used to determine the time to boil under the specified scenarios.

Additional heat input required to boil the water to the top of the spent fuel racks.

The volume of the pools that is above the top of the racks was determined. Using this volume, the heat input required to change this volume of water from a saturated liquid to a saturated vapor was calculated using standard heat transfer equations.

Additional time to boil down to the top of the racks.

The heat input required to boil the water down to the top of the spent fuel racks was divided by the heat load in the pools to yield the additional time after the start of boiling required to reach the top of the racks. Best estimate heat loads were used to determine the time to boil off the water under the postulated scenarios.

Total time from initiation until the water has boiled down to the top of the racks.

The previously calculated time to boil plus the additional time to reach the top of the racks once boiling begins provides the total time available for the operators to reestablish either cooling or makeup to the pools in order to prevent uncovery of the fuel in the spent fuel pools.

Makeup flow rate required to offset boiling.

The rate (in gallons per minute) that the water is being boiled off was determined by dividing the previously calculated volume of water above the racks by the additional time required to boil the water down to the top of the racks. This boiloff rate is also the amount of makeup required to maintain pool level constant under boiling conditions.

Attachment E

Data Sources for Input Values and Initial Conditions

Nominal spent fuel pool level (284.5 ft)	CAR-2165-G-024, System Description SD-116 (Fuel Pool Cooling and Cleanup System), FSAR and DBD
Elevation of top of racks (260.08 ft)	CAR-2165-G-024, System Description SD-116 (Fuel Pool Cooling and Cleanup System), FSAR and DBD
Bottom of pools (246 ft)	CAR-2165-G-024, System Description SD-116 (Fuel Pool Cooling and Cleanup System), FSAR and DBD
Bottom of gates (260 ft)	CAR-2165-G-024, FSAR and DBD
Best estimate of spent fuel pool temperature at initiation (95°F)	Derived from historical spent fuel pool temperatures between August 1999 and September 2000 as recorded by the plant Emergency Response Facility Information System ("ERFIS") computer.
Pools A and B beginning of cycle heat load (25.0 MBTU/hr)	Core shuffle refueling alignment heat load from calculation SF-0040, "Spent Fuel Pools C and D Activation Project Thermal-Hydraulic Analysis." This heat load is the future expected heat load when pools A and B would be essentially full. This number is expected to be conservative since the number calculated in ESR 00-00046 for cycle 10 is less than 16.84 MBTU/hr.
Pools A and B base heat load at the end of an operating cycle (13.3 MBTU/hr)	Obtained from calculation SF-0040, "Spent Fuel Pools C and D Activation Project Thermal-Hydraulic Analysis."
Pools C and D initial licensed limit heat load (1.0 MBTU/hr)	Technical Specification limit in the pending License Amendment Request (HNP-98-188).

Pools C and D maximum end of life heat load (15.6 MBTU/hr)	ESR 95-00442 Action Item 2.
Specific Volume of water at 95°F (0.016115 cu ft/lb)	IAPWS 97 Formulation
Specific Volume of water at 212°F (0.016714554 cu ft/lb)	IAPWS 97 Formulation
Enthalpy of water at 95°F (63.0459 BTU/lb)	IAPWS 97 Formulation
Enthalpy of water at 212°F under saturated water conditions (180.1802 BTU/lb)	IAPWS 97 Formulation
Enthalpy of water at 212°F under saturated vapor conditions (1,150.2889 BTU/lb) – IAPWS 97 Formulation	IAPWS 97 Formulation
Density of water (7.48 gal/cu ft)	standard conversion formula
Pool A rack layout	Drawing CAR-2168-G-0124 S01 (Fuel Rack Arrangement Harris Fuel Pool "A")
Pool B rack layout	Drawing CAR-2168-G-0116 S01 (Fuel Rack Arrangement Harris Fuel Pool "B")
Pool A rack weights	Harris Plant procedure MMM-020
Pool B rack weights	Harris Plant procedure MMM-020
Pool C rack layout and weights	Holtec drawing 1994
Pool D rack layout and weights	Holtec drawing 1993
Specific weight of a spent fuel rack (0.29 lb/cu in)	Mark's Standard Handbook (stainless steel)
Volume of a BWR fuel assembly (1.164 cu ft)	Engineering Service Request 95-00584

Volume of a PWR fuel assembly (2.63 cu ft)	Engineering Service Request 95-00584
Pool A dimensions – (38 ft x 13 ft)	Drawing CAR-2165-G-022
Pool B dimensions – (50 ft x 27 ft)	Drawing CAR-2165-G-022
Pool C dimensions – (50 ft x 27 ft)	Drawing CAR-2165-G-022
Pool D dimensions – (32 ft x 20 ft)	Drawing CAR-2165-G-022
Unit 1/4 transfer canal dimensions (38 ft x 9 ft)	Drawing CAR-2165-G-022
Unit 2/3 transfer canal dimensions (38 ft x 9 ft)	Drawing CAR-2165-G-022
Main transfer canal (288 ft x 3 ft)	Drawing CAR-2165-G-022

Attachment F

Summary Results of Heatup Calculations for Analyzed Scenarios

Pools	Time to reach boiling temperature	Additional time for water level to reach top of racks	Total time	Makeup required to offset boiling
A and B (Beginning of cycle)	20.57 hours	7.21 days	8.07 days	53.70 gpm
A and B (End of cycle)	38.67 hours	13.56 days	15.17 days	28.57 gpm
C and D (1 MBTU/hr heat load)	384.66 hours	99.99 days	116.02 days	2.15 gpm
C and D (15.6 MBTU/hr heat load)	34.42 hours	8.80 days	10.23 days	33.64 gpm

	A	B	C	D	E	F
1			**Supporting Analysis for NRC Specific Interrogatory #7**			
2						
3			Request: Calculate the best estimate time to reach boiling temperature and then additional time for water level to boil down to the top of the racks			
4			Also calculated is the amount of makeup water required to offset the boiloff rate (in gpm)			
5						
6			**Assumptions:**			
7						
8			1 Gates 1, 3 & 4 are removed. Gate 2 is installed.			
9			Thus SFP A, SFP B, 1/4 transfer canal and main transfer canal are interconnected. This is normal lineup.			
10			2 Gate 7 is removed and gates 5 & 6 are installed for SFP C 1 MBTU/hr scenario. Thus SFP C and 2/3 transfer canal are interconnected for cooling.			
11			SFP D is not considered for 1MBTU/hr scenario since all racks and fuel will be in SFP C.			
12			3 Gates 7 & 9 are removed and gates 5 & 6 are installed for SFP C&D 15.6MBTU/hr scenario.			
13			Thus SFP C, SFP D and 2/3 transfer canal will be interconnected for cooling. This will be the normal lineup when SFP C&D are activated.			
14			4 Water volume in main transfer canal and cask unloading pool are not considered in this analysis.			
15			5 No credit is taken for any SFP cooling or makeup after initiation of the loss of cooling occurs.			
16			6 Beginning Cycle heat load for SFP A&B assumes base heat load from SF-40 plus 1/3 freshly discharged core after start up from a refueling outage			
17			7 No credit is taken for heat transfer to liner, concrete structure or atmosphere.			
18						
19						
20			**Inputs:**			
21						
22			Nominal water level	284.50	ft.	
23			Top of Racks	260.08	ft.	
24			Bottom of Pools	246.00	ft.	
25			Bottom of Gates	260.00	ft.	
26			Best estimate - SFP temp at initiation	95	deg. F	
27						
28			Pool A&B heat load	25,000,000	BTU/hr	Beginning of cycle
29			Pool A&B heat load	13,300,000	BTU/hr	Base heat load (end of cycle)
30			Pool C&D heat load	1,000,000	BTU/hr	Initial Licensed Limit
31			Pool C&D heat load	15 661,901	BTU/hr	End of Life heat load
32						
33			Water Properties:			
34			Specific Volume @ 95F (initation temperature)	0.016115214	cu ft/lb	
35			Specific Volume @ 212F (saturated liquid)	0.016714554	cu ft/lb	
36			Enthalpy @ 95F (initiation temperature)	63.0459	BTU/lb	
37			Enthalpy @ 212F (saturated liquid)	180.1802	BTU/lb	
38			Enthalpy @ 212F (saturated vapor)	1,150.2889	BTU/lb	
39				7.48	gal/cu ft	
40						
41			Rack & fuel volume:			
42			Pool A	1,675.85	cu ft	
43			Pool B	5,627.63	cu ft	
44			Pool C	6,449.04	cu ft	
45			Pool D	3,007.77	cu ft	
46						
47			Pool A water volume - total	17,343.15	cu ft	38ft x 13ft x (284.5 - 246)ft minus rack/fuel volume
48			Pool A water volume - to top of racks	12,063.48	cu ft	38ft x 13ft x (284.5 - 260.08)ft
49			Pool B water volume - total	46,347.37	cu ft	50ft x 27ft x (284.5 - 246)ft minus rack/fuel volume
50			Pool B water volume - to top of racks	32,967.00	cu ft	50ft x 27ft x (284.5 - 260.08)ft
51			Pool C water volume - total	45,525.96	cu ft	50ft x 27ft x (284.5 - 246)ft minus rack/fuel volume
52			Pool C water volume - to top of racks	32,967.00	cu ft	50ft x 27ft x (284.5 - 260.08)ft
53			Pool D water volume - total	21,632.23	cu ft	32ft x 20ft x (284.5 - 246)ft minus rack/fuel volume
54			Pool D water volume - to top of racks	15,628.80	cu ft	32ft x 20ft x (284.5 - 260.08)ft
55			1/4 transfer water volume to bottom of gates	8,379.00	cu ft	38ft x 9ft x (284.5 - 260)ft
56			2/3 transfer water volume to bottom of gates	8,379.00	cu ft	38ft x 9ft x (284.5 - 260)ft
57			Main transfer canal water volume	21,168.00	cu ft	288ft x 3ft x (284.5 - 260)ft
58						
59						
60						
61			**Pools A & B at the beginning of the cycle**			
62						
63			**Time to boil for A&B - beginning of cycle:**			
64			Volume available for saturation	72,069.52	cu ft	total volume of SFP A, SFP B & 1/4 canal
65			Mass	4,390,493.25	lb	total volume / specific volume
66			Heat input required to reach boiling	514,277,884.92	BTU	mass x (enthalpy @ initiation - enthalpy @ saturated liquid)
67			**Time required to boiling**	20.57	hr	heat input / heat load at beginning of cycle
68						
69			**Additional time to boil down to top of racks - beginning of cycle:**			
70			Volume available to top of racks	74,577.48	cu ft	volume to top of racks for SFP A, SFP B, main canal & 1/4 canal
71			Mass	4,461,828.86	lb	volume to top of racks / specific volume
72			Heat input required to reach top of racks	4,328,459,070.68	BTU	mass x (enthalpy @ saturated vapor - enthalpy @ saturated liquid)
73			**Time to reach top of racks**	173.14	hr	heat input / heat load
74			or	7.21	days	
75						
76			**Total time from initiation until top of racks:**			
77			**Time to boil + time to top of racks**	193.71	hr	
78				8.07	days	
79						
80			**Makeup required to offset boiling:**			
81			Volume to top of racks	74,577.48	cu ft	volume to top of racks for SFP A, SFP B, main canal & 1/4 canal
82			Volume to top of racks	557,839.55	gallons	convert cu ft to gallons (x by 7.48)
83			Time to boil to top of racks	173.14	hr	from above
84			**Make up flow rate required**	53.70	gal/min	volume to top of racks / time to top of racks after boiling begins
85						

	A	B	C	D	E	F
86						
87						
88		Pools A & B at the end of the cycle				
89						
90		Time to Boil for A&B - End of cycle:				
91			Heat input required to reach boiling	514,277,884.92	BTU	calculated above
92			Time required to boiling	38.67	hr	heat input / heat load at end of cycle
93						
94		Additional time to boil down to top of racks - End of cycle:				
95			Heat input required to reach top of racks	4,328,459,070.68	BTU	calculated above
96			Time to reach top of racks	325.45	hr	heat input / heat load at end of cycle
97			or	13.56	days	
98						
99		Total time from initiation until top of racks:				
100			Time to boil + time to top of racks	364.12	hr	
101			or	15.17	days	
102						
103		Makeup required to offset boiling:				
104			Volume to top of racks	74,577.48	cu ft	volume to top of racks for SFP A, SFP B & 1/4 canal
105			Volume to top of racks	557,839.55	gallons	convert cu ft to gallons (x by 7.48)
106			Time to boil to top of racks	325.45	hr	from above
107			Make up flow rate required	28.57	gal/min	volume to top of racks / time to top of racks after boiling begins
108						
109						
110						
111						
112		Pool C at the initial licensed heat load of 1 MBTU/hr				
113						
114		Time to Boil for C - Initial Licensed Heat Load:				
115			Volume available for saturation	53,904.96	cu ft	total volume of SFP C & 2/3 canal
116			Mass	3,283,907.43	lb	total volume/specific volume
117			Heat input required to reach boiling	384,658,158.96	BTU	mass x (enthalpy @ initiation - enthalpy @ saturated liquid)
118			Time required to boiling	384.66	hr	heat input / heat load at initial licensed limit
119			or	16.03	days	
120						
121		Additional time to boil down to top of racks - Initial Licensed Heat Load:				
122			Volume available to top of racks	41,346.00	cu ft	volume to top of racks for SFP C & 2/3 canal
123			Mass	2,473,652.58	lb	volume to top of racks / specific volume
124			Heat input required to reach top of racks	2,399,711,933.64	BTU	mass x (enthalpy @ saturated vapor - enthalpy @ saturated liquid)
125			Time to reach top of racks	2,399.71	hr	heat input / heat load
126			or	99.99	days	
127						
128		Total time from initiation until top of racks:				
129			Time to boil + time to top of racks	2,784.37	hr	
130			or	116.02	days	
131						
132		Makeup required to offset boiling:				
133			Volume to top of racks	41,346.00	cu ft	volume to top of racks for SFP C & 2/3 canal
134			Volume to top of racks	309,268.08	gallons	convert cu ft to gallons (x by 7.48)
135			Time to boil to top of racks	2,399.71	hr	from above
136			Make up flow rate required	2.15	gal/min	volume to top of racks / time to top of racks after boiling begins
137						
138						
139						
140		Pool C&D at maximum end of life heat load of 15.6 MBTU/hr				
141						
142		Time to Boil for C&D - End of Life Heat Load:				
143			Volume available for saturation	75,537.19	cu ft	total volume of SFP C, SFP D & 2/3 canal
144			Mass	4,601,749.81	lb	total volume/specific volume
145			Heat input required to reach boiling	539,022,686.79	BTU	mass x (enthalpy @ initiation - enthalpy @ saturated liquid)
146			Time required to boiling	34.42	hr	heat input / heat load at maximum end of life heat load
147			or	1.43	days	
148						
149		Additional time to boil down to top of racks - End of Life Heat Load:				
150			Volume available to top of racks	56,974.80	cu ft	volume to top of racks for SFP C, SFP D & 2/3 canal
151			Mass	3,408,693.98	lb	volume to top of racks / specific volume
152			Heat input required to reach top of racks	3,306,803,741.03	BTU	mass x (enthalpy @ saturated vapor - enthalpy @ saturated liquid)
153			Time to reach top of racks	211.14	hr	heat input / heat load
154			or	8.80	days	
155						
156		Total time from initiation until top of racks:				
157			Time to boil + time to top of racks	245.55	hr	
158			or	10.23	days	
159						
160		Makeup required to offset boiling:				
161			Volume to top of racks	56,974.80	cu ft	volume to top of racks for SFP C, SFP D & 2/3 canal
162			Volume to top of racks	426,171.50	gallons	convert cu ft to gallons (x by 7.48)
163			Time to boil to top of racks	211.14	hr	from above
164			Make up flow rate required	33.64	gal/min	volume to top of racks / time to top of racks after boiling begins
165						

EXHIBIT 5

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

<u>Before the Atomic Safety and Licensing Board</u>

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

<u>AFFIDAVIT OF ERIC A. MCCARTNEY</u>

COUNTY OF WAKE	)
	) ss:
STATE OF NORTH CAROLINA	)

I, Eric A. McCartney, being sworn, do on oath depose and say:

1. I am a resident of the State of North Carolina. I am employed by Carolina Power & Light Company ("CP&L") and work at the Harris Nuclear Plant ("HNP" or "Harris Plant" or "Harris") in the Licensing Department. Presently, I am the Supervisor, Licensing/Regulatory Programs, and am responsible for managing regulatory interfaces for Harris. My business address is 5413 Shearon Harris Road, New Hill, North Carolina 27562-0165.

2. I was graduated from the University of Maryland in 1995 with a B.S. in Nuclear Science. Between 1974 and 1995, I have held several positions of increasing responsibility in nuclear power operations, first in the United States Navy, then

with CP&L. I obtained my Reactor Operator license in 1988 and have been a licensed Senior Reactor Operator since 1991. Since graduation from the University of Maryland, I have been employed by CP&L in supervisory positions, first as Superintendent – Shift Operations, responsible for shift activities including radioactive waste and makeup water systems at Harris. In 1997, I was appointed Harris Superintendent – Work Control, responsible for directing day-to-day licensed activities as the senior licensed operator onsite. Between January 1999 and April 2000, I was an Operations Evaluator for the Institute of Nuclear Power Operations ("INPO") performing evaluations of nuclear power plant operations. I also performed World Association of Nuclear Operators ("WANO") peer reviews and assistance visits at Donald C. Cook Nuclear Power Station and Fort Calhoun Nuclear Station. In April 2000, I was appointed to my current position of Supervisor – HNP Licensing/Regulatory Programs, where I am responsible for preparing regulatory reports, supporting plant operations, and participating on the Plant Nuclear Safety Committee. I have also served on the Technical Support Center ("TSC") staff as the Site Emergency Coordinator, responsible for operating decisions during emergency conditions. My resume is included as Attachment A to this affidavit.

3. The purpose of this affidavit is to set forth facts on which CP&L relies in establishing that numerous, diverse sources of water and methods of delivery exist for establishing makeup to the Harris spent fuel pools. First, I describe the Harris

spent fuel pool physical arrangement, systems configurations, and plant equipment associated with normal and alternate makeup to the spent fuel pools. Second, I discuss the methods available for supplying makeup water to the Harris spent fuel pools and identify the Harris procedures, controls, conditions, and equipment that establish the viability of each method. Third, I describe the TSC, its functions and personnel, and how the Severe Accident Management Guidelines ("SAMGs") are used to assist the operating staff in responding to emergency conditions outside of existing procedures. Finally, I provide my conclusions on the ability of Harris operators to restore makeup water to the spent fuel pools under emergency conditions.

HARRIS PLANT SPENT FUEL POOLS, ASSOCIATED SYSTEMS AND AVAILABLE EQUIPMENT

4. As a consequence of my extensive experience as a licensed operator and manager at Harris, I am familiar with the physical layout, system configurations, equipment installations, operations, and operating procedures for Harris, as they relate to normal and alternate operation of the fuel handling building, the spent fuel pools, and associated water sources, support systems and equipment.

5. As Harris was originally envisioned as a four unit facility with a shared fuel handling building, the fuel handling building was designed and constructed with four separate pools capable of storing spent nuclear fuel. Spent fuel pools A and B were originally intended to support Harris Units 1 and 4. Spent fuel pools C

and D were originally intended to support Harris Units 2 and 3. In addition, the fuel handling building contains a cask unloading pool, which can be connected to any spent fuel pool through transfer canals.

6. The layout of the Harris fuel handling building is illustrated and described in detail in HNP FSAR, section 9.1. Each spent fuel pool and the cask unloading pool are interconnected by a main transfer canal, oriented in a north-south direction, and two fuel transfer canals, oriented east-west. The spent fuel pools and transfer canals contain sufficient amounts of water to facilitate safe fuel handling and storage activities. The spent fuel pools, transfer canals, and cask unloading pool contain openings for the underwater movement of fuel assemblies between the pools and transfer canals. These openings also allow the communication of water between the pools and transfer canals. Removable bulkhead gates are installed in the openings when there is a need to isolate a particular pool or canal from the others. The isolation function of the bulkhead gates is provided by stainless steel structural components and inflatable seals, which are installed around the sides of the gates that fit into slots in the pool and canal openings. The seals are normally inflated using instrument air supplied at the installed gate location.

7. The normal configuration of the spent fuel pools (*i.e.*, the configuration expected 99% of the time an on annual basis) is with open communication (*i.e.*, the gates removed) between spent fuel pools A and B and the connecting transfer canal.

The gate between the transfer canal and the main fuel transfer canal is also normally removed. The cask unloading pool is normally connected to the main fuel transfer canal. Spent fuel pools C and D are currently normally isolated from spent fuel pools A and B by a gate installed between the main fuel transfer canal and pools C and D. This alignment is illustrated in Attachment B. Once placed in operation, spent fuel pools C and D and the associated fuel transfer canal will normally be connected to the cask handling pool. Spent fuel pools A and B and their shared transfer canal will be connected to the main fuel transfer canal. This alignment is illustrated in Attachment C.

8. The original Harris design included a spent fuel pool cooling and cleanup system ("SFPCCS") to service spent fuel pools A and B, and a separate SFPCCS to service spent fuel pools C and D. The SFPCCS for spent fuel pools A and B is in service. The SFPCCS for spent fuel pools C and D was not completed, but will be finished and placed in service to support spent fuel operations pursuant to the pending license amendment request.

9. The purpose of the SFPCCS is to maintain water quality in the spent fuel pools, transfer canals, cask loading pool and the reactor cavity, and remove residual heat generated in the stored spent fuel. The SFPCCS consists of a cooling system and a cleanup system. The major system components are the fuel pool heat exchangers, fuel pool demineralizer, fuel pool cooling pumps, filters, skimmers, water purification pumps, valves, piping, fuel pool gates, strainers,

instrumentation and system controls. The SFCCS is comprised of two separate cooling loops, each with 100% capacity and independence. The fuel pool cooling pumps are powered from train separated power sources with the capability of being connected to the emergency diesel generator should a loss of offsite power occur. The normal source of water for the system is the Refueling Water Storage Tank ("RWST"), which has a capacity of 469,000 gallons and is maintained above 436,000 gallons at all times during reactor operation. Attachment D is the system description, SD-116, from Volume 6 of the Harris Plant Operating Manual, which provides a more detailed description of the SFPCCS. Attachment E is a simplified schematic of the system.

10. The demineralized water system ("DWS") is designed to process filtered water from the filtered water makeup system to produce demineralized water sufficient for the expected demands during startup and operation of various Harris plant systems, including the reactor coolant system. The DWS is capable of supplying normal makeup needs with additional capacity for maintaining level in the condensate storage tank, refueling water storage tank, and reactor makeup water storage tank, which provide makeup water capacity to bring the plant to a shutdown condition during accident conditions. The major system components include: the demineralized raw water feed pumps, carbon filters, cation, anion and mixed bed demineralizers, a vacuum degassifier, degassified water transfer pumps, a demineralized water storage tank with a capacity of 500,000 gallons,

and demineralized water storage tank transfer pumps. Attachment F is the system description, SD-143, from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the DWS. Attachment G is a simplified schematic of the system.

11. The RWST has two functions. First, it provides a borated water source for the charging safety injection pumps, containment spray pumps and residual heat removal pumps for injection into the reactor vessel during accident conditions that threaten core uncovery. The RWST is also a source of water to fill the fuel transfer canals for refueling operations. The RWST has a capacity of 469,000 gallons and is maintained greater than 92% at all times while the plant is operating. Technical Specifications require the engineered safety features actuation system to swap safety injection pump suction from the RWST to the containment sump at 23.4% RWST level to prevent possible safety injection pump damage. This leaves approximately 100,000 gallons of water available for use by other systems following safety injection. Attachment H is the system description, SD-112, from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the RWST. Attachment I is a simplified schematic of the RWST and connecting systems.

12. The normal service water system ("NSW") circulates water from the cooling tower and cooling tower makeup system through plant auxiliary components and back to the cooling tower. The NSW provides cooling water to the ESW headers

during normal operations. The key components of the system include two (2) NSW pumps and interconnecting piping. The pumps are powered from non-safety auxiliary busses. The sources of water for NSW are the main and auxiliary reservoirs. Attachment J is the system description, SD-139 from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the NSW System. Attachment K is a simplified schematic of the NSW System.

13. The emergency service water ("ESW") system circulates water from the ultimate heat sink ("UHS") through the plant components required for safe shutdown of the reactor following an accident and returns the water to the UHS. The ESW system provides an emergency source of water for the auxiliary feedwater system, essential services chilled water system and fire protection system. Key system components include two (2) ESW pumps and two (2) ESW booster pumps, are powered from safety-related buses. The sources of water for the ESW system are the main and auxiliary reservoirs. Attachment J is the system description, SD-139, from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the ESW system. Attachment L is a simplified schematic of the system.

14. The reactor makeup water storage tank ("RMWST") serves as a storage volume for makeup water to nuclear steam supply system, specifically the component cooling water ("CCW") system and reactor coolant system ("RCS"). The RMWST has a capacity of 85,000 gallons. Attachment M is the system

description, SD-102, from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the RMWST. Attachment N is a simplified schematic of the RMWST and connecting systems.

15. The purpose of the fire protection system is to ensure the capability to shut down the reactor safely, maintain it in a safe shutdown condition, and limit the radioactive release to the environment in the event of a fire. It also serves to minimize both the probability and the consequences of fires, thereby protecting plant personnel and plant related equipment and property. Two fire pumps are provided, one motor driven pump powered from a non-safety power supply and one diesel engine driven pump, each with a capacity of 2500 gallons per minute. Both fire pumps draw suction from the Auxiliary Reservoir. Attachment O is the system description, SD-149, from Volume 6 of the Harris Plant Operating Manual that provides a more detailed description of the Fire Protection System. Attachment P is a simplified schematic of the system.

16. Harris has agreements with the Holly Springs and Apex Fire Departments to provide assistance in emergency situations. The Apex fire station is approximately three miles from the Harris site. Generally, Holly Springs and Apex respond to the site in 15 to 30 minutes from the phone call requesting assistance. Additional emergency resources are available from Raleigh and Sanford fire departments, with which Harris has also established working relationships. These fire departments are familiar with Harris because they

participate in annual drills requiring off-site fire department response as part of the Harris Emergency Plan.

17. Harris Fire Brigade members are provided with a full set of structural fire fighting turnout gear and Self-Contained Breathing Apparatus ("SCBA") suitable for performing actions in high temperature and high humidity conditions. This gear includes boots, bib overalls, coat, hood and helmet. Harris turnout gear is generally manufactured from Nomex™ and Gortex™ materials, which are non-combustible and vent moisture during use.

18. Plant Fire Brigade members undergo initial and annual re-qualification LIVE Fire Training using turnout gear and SCBA. This training includes structural fire training (entry and extinguishing a actual structural fire) in a building. Fires during these training evolutions typically involve temperatures in the fire room well above the 195° F that is anticipated in the Harris fuel handling building during postulated spent fuel pool boiling. Temperatures in the range of 300° F would not be unusual during Fire Brigade training exercises. Fire fighting with a fire hose normally produces high humidity conditions and Fire Brigade members are trained to perform under such conditions. Additionally, Fire Brigade members conduct periodic drill in various plant locations to exercise their ability to use fire fighting equipment and techniques.

19. Emergency lighting is installed and regularly maintained in locations where operator actions may be required following a loss of normal lighting. In

particular, the plant areas containing valves and equipment necessary to provide normal and alternate fuel pool cooling and make up contain normal and emergency lighting, either of which is adequate to enable an operator to perform the actions required to establish flow to the fuel pools. In addition, portable, battery-powered flashlights and lanterns are available in each emergency facility and the control room. These storage locations are periodically inventoried to ensure the equipment is available and working. Normal and emergency lighting is more fully described in system description SD-158, Plant Lighting (Attachment Q).

20. Ladders are strategically located in designated ladder storage areas throughout the fuel handling and reactor auxiliary buildings. Ladder locations are marked with placards and a log is maintained in the operator's work area showing the location of ladders in the plant. These ladders are staged for the specific purpose of providing operators access to elevated valves and equipment, and operators are familiar with their locations.

21. Auxiliary operators are trained to operate locked doors and valve locking devices under normal and emergency conditions. The doors in the fuel handling building and reactor auxiliary building do not require electrical power to be operated manually. Power provides only alarm and indication for the security system. Each operator carries keys that operate plant doors, as well as keys for locked equipment, such as valves and tool boxes. Additional keys of both types are

maintained in the main control room. In addition, following the activation of the Emergency Plan, or on a loss of site power, security personnel are tasked to assist operating staff with access to locked areas. Finally, bolt cutters and torches can be obtained from the site tool room located in the service building. Operators and security personnel are trained and familiar with operating plant doors and equipment under normal and emergency conditions.

METHODS FOR SUPPLYING MAKEUP WATER TO THE HARRIS SPENT FUEL POOLS

22. There are numerous normal and alternate methods for supplying makeup water to the Harris spent fuel pools. To the best of my knowledge and belief, each of the ten (10) methods described below is individually capable of delivering makeup water to the spent fuel pools at a rate in excess of the highest evaporation or boil-off rates calculated for the beyond-design-basis accident with containment bypass scenario described in Contention EC-6.

23. In each method described below, makeup water will be available to each spent fuel pool regardless of the makeup water discharge location or the bulkhead gate configuration. Makeup is available to spent fuel pool A and B (and spent fuel pools C and D when they are placed in service) in the normal bulkhead gate configuration by open communication through the transfer canals. If makeup is required when a spent fuel pool is isolated by one or more installed bulkhead gates, makeup water can be made available simply by depressurization of the inflated gate seals. With the seals depressurized, makeup water will communicate

into, and equalize the levels of, the spent fuel pools, transfer canals, and cask unloading pool. The seals can be deflated locally by bleeding the pressure off through a vent valve. In the event the gate seals cannot be depressurized, makeup water from a filling pool or transfer canal will overflow the installed gates and fill the other pools and canals.

24. In addition to the ten (10) methods described below, following installation of the plant modifications associated with spent fuel pools C and D, a completely redundant spent fuel pool cooling system, purification system, and skimmer system will be installed in the north end of the fuel handling building. This will provide four (4) additional redundant delivery locations for operators to align existing makeup water sources to the spent fuel pools, transfer canals, and cask loading pool. Procedure OP-116 will be revised to reflect the additional redundant makeup water pathways before adding spent fuel to spent fuel pool C.

Normal Makeup

25. Normal spent fuel pool makeup is accomplished by OP-116, Revision 17, section 8.4, "Makeup to SFP B with Demineralized Water with the Purification System in Service" (Attachment R). To initiate normal makeup, plant operators have to open only a single manual valve, 1SF-201, located on the 216-foot elevation at the south end of the fuel handling building. This action aligns the demineralized water system to the spent fuel purification system and delivers makeup water directly to spent fuel pool B. As described above, makeup is available to spent

fuel pool A (and spent fuel pools C and D when they are placed in service) in the normal bulkhead gate alignment by open communication through the transfer canals and to other pools by either deflating the gate seals or overflowing the gates. The source of water for this mode of makeup is the demineralized water storage tank, which has a capacity of 500,000 gallons. The flow rate is approximately 100 gallons per minute. An operator can reach the valve needed to align this flow path within approximately 15 minutes from any location in the reactor auxiliary building ("RAB") or fuel handling building ("FHB") and can initiate flow within another 5 minutes. The procedural reference for this method is OP-116, Revision 17. Once the SFPCCS for spent fuel pools C and D is placed in service, normal makeup to pools C and D will be initiated by opening a single manual valve, 2SF-201, located on the 216-foot elevation at the north end of the fuel handling building.

<u>Alternate Makeup Method No. 1</u> – Demineralized Water Tank to Spent Fuel Pool Purification System

26. This method aligns the demineralized water tank to the spent fuel pool purification system and delivers water to spent fuel pool A, spent fuel pool B, the interconnecting transfer canal, the cask loading pool, or all of these locations. To provide makeup water to spent fuel pools A and B, the operators manually opens 1SF-201 and any one of the following valves 1SF-26, -27, -28, -29, or –192. One of these valves is located on the 216 foot elevation of the FHB, two are on the 236 foot elevation of the FHB, and one is on the 261 foot elevation of the FHB. To

align makeup water to the cask loading pool an additional five (5) valves, 1SF-206, 2SF-205, 2SF-141, 188, and 203, are opened. All of these valves are located in the FHB on the 216 foot elevation. After establishing the desired valve lineup, the operator closes power supply breakers 1-4A1021-1D and 1-4B1021-5E for the spent fuel pool purification pumps on the 261-foot elevation of the FHB. Once energized, the operator starts the purification pump from one of two locations (*i.e.*, the FHB operating deck or the FHB 236-foot elevation). The source of water is the demineralized water storage tank with a capacity of 500,000 gallons with a flow rate of approximately 100 gallons per minute. An operator can access these valves within 15 minutes and initiate flow in approximately 30 minutes. The procedural reference for this method is OP-116, Revision 17, section 8.5. When spent fuel pools C and D and the associated SPFCCS are placed in service, a redundant alignment to implement this alternate method will be available by opening 2SF-201 and any one of the following valves, 2SF-26, -27, -28, -29 or -192.

<u>Alternate Makeup Method No. 2</u> – Refueling Water Storage Tank (RWST) to Spent Fuel Pool Purification System

27. This method aligns the RWST to spent fuel pools A and B, the interconnecting transfer canal, the cask loading pool, or all of these locations simultaneously. To align this flow path, an operator manually shuts 1SF-202 on FHB 216 and opens one or all of the following valves: 1SF-26 or -27 to spent fuel pool A; 1SF-28 or -29 to spent fuel pool B; or 1SF-192 to the south transfer canal. One of these

valves is on the FHB 216 foot elevation and the remainder are on the FHB 236 foot elevation. The operator then opens 1CT-23 on the RAB 236 foot elevation. After establishing the desired valve lineup, the operator closes power supply breakers 1-4A1021-1D and 1-4B1021-5E for the spent fuel pool purification pumps on the FHB 261-foot elevation. Once energized, the operator starts the purification pump from one of two locations (*i.e.*, the FHB operating deck or the FHB 236-foot elevation). The source of this flow path is the RWST with a capacity of 469,000 gallons. If the RWST is approximately 50% or more full, as required during plant operations, this flow path will result in gravity flow to the spent fuel pools, transfer canal, or cask loading pool with an expected flow rate of up to 100 gallons per minute. When spent fuel pools C and D and the associated SFPCCS are placed in service, a redundant method to implement this flow path will be available by aligning 2SF-202, -26, -27, -28, -29, and 192. The Unit 1 RWST will be the source of water through 1CT-23. An operator can access these valves within approximately 15 minutes and initiate flow in approximately 30 minutes. The procedural reference for this method is OP-116, Revision 17, section 8.5.

<u>Alternate Makeup Method No. 3</u> – Demineralized Water System to Spent Fuel Pool Skimmer System

28. This method aligns the demineralized water system to the spent fuel pool skimmer system and delivers makeup water to spent fuel pool A, spent fuel pool B, or the

transfer canal. The method requires the skimmer system to be in service, which is normally the case. To align this flow path, the operator simply opens valve 1DW-527 on the 236-foot elevation of the fuel handling building. The source of water is the demineralized water storage tank with a capacity of 500,000 gallons. The flow rate is approximately 100 gallons per minute. An operator can access this valve and initiate flow in approximately 5 minutes. The procedural reference for this method is OP-116, Revision 17, section 8.6.

<u>Alternate Makeup Method No. 4</u> – Refueling Water Storage Tank (RWST) to Spent Fuel Pool Cooling System

29. This method aligns the RWST to the suction of the spent fuel pool cooling pumps and delivers water to the spent fuel pool A/B transfer canal, the main fuel transfer canal, or the cask loading pool. To align this flow path the operator manually aligns eleven (11) valves. Eight (8) of the valves, 1SF-1, -5, -9, -10, -21, -23, -24, and -25, are located on the FHB 236-foot elevation; 1SF-193 is located on the FHB 216-foot elevation, and two (2) valves, 3BR-378 and 1CT-23, are located on the RAB 236-foot elevation. The source of water is the RWST with a capacity of 469,000 gallons. If the RWST level is above approximately 50% full (as is required during reactor operation) then the transfer canal or cask loading pool will fill due to gravity with a flow rate of up to approximately 500 gallons per minute. The spent fuel pool cooling pump is then started from the main control room to distribute water to all the pools. An operator can access these valves within approximately 15 minutes and initiate flow in approximately 30 minutes. The

procedural reference for this method is OP-116, Revision 17, section 8.12. When spent fuel pools C and D and the associated SFPCCS are placed in service, a redundant flow path will be available by aligning 2SF-1, -5, -9, -10, -21, -23, -24, -25, and -193. The Unit 1 RWST is the source through valve 1CT-23.

<u>Alternate Makeup Method No. 5</u> – Emergency Service Water (ESW) System to Spent Fuel Pool Cooling System

30. This method aligns the ESW system to the spent fuel pool cooling system. To align this flow path, the operator connects a jumper between spent fuel pool cooling system connection valves, 1SW-1239(269) and 1SF-76, on the RAB 236-foot elevation. A toolbox is staged locally which contains all the necessary hose and fittings to install the jumper. The operator then opens the two valves to initiate flow. The source of water is Harris Lake, which provides a virtually unlimited supply of makeup water at a flow rate is approximately 50 to 75 gallons per minute. The operator can align this flow path within 30 minutes, as all the tools and equipment necessary to align this path are with 50 feet of the valves. The procedural reference for this method is OP-116, Revision 17, section 8.13.

<u>Alternate Makeup Method No. 6</u> – Reactor Makeup Water Storage Tank (RMWST) to Spent Fuel Pool Purification System

31. This method aligns the RMWST to the spent fuel pools. To connect the RMWST to the spent fuel pool purification system, an operator must manually align four (4) valves: 1SF-194 on FHB elevation 216, and 3PM-83, 1PM-81, and 1PM-150 on RAB elevation 261. The source of water is the RMWST with a capacity of

80,000 gallons. The flow rate is 75 to 100 gallons per minute. An operator can access these valves within approximately 15 minutes and initiate flow in approximately 30 minutes. The procedural reference for this method is OP-116, Revision 17, section 8.26.

<u>Alternate Makeup Method No. 7</u> – Fire Protection System to Spent Fuel Pool or Transfer Canals

32. The plant fire protection system draws water from the Harris Lake via a motor driven fire pump or a redundant diesel driven fire pump. There are seven (7) fire hose stations on the operating deck of the fuel handling building, each equipped with a 1 1/2 inch fire hose and connected to a normally pressurized fire water header. Several of these stations, at least one on each level of the building, are seismically qualified. The fire header is maintained pressurized by the fire pump. If the motor driven pump fails, a diesel driven fire pump will automatically start as a backup. To begin filling the spent fuel pools or transfer canal, an operator simply aims a fire hose at the spent fuel pool and opens the hose station valve. Each fire hose can deliver approximately 125 gallons per minute. The fire hoses can be attached to a railing in the fuel handling, precluding the need to station an operator on the fuel handling deck. An operator can access these valves and equipment and initiate flow in approximately 5 minutes.

<u>Alternate Makeup Method No. 8</u> – Normal Water Service ("NSW") System to Spent Fuel Pool or Transfer Canals

33. This method uses water from the NSW system to fill an accessible spent fuel pool

or transfer canal through a hose. The NSW system extends into the waste processing building, where an operator connects a 1-inch drain hose to a NSW system drain valve, 1SW-817, at the FHB 261-foot elevation stair well. Tools and hose can be obtained from the tool room in the service building. The operator then runs approximately 300 feet of hose from the valve to deliver makeup water to spent fuel pool A. A one inch drain hose is expected to provide a flow rate of approximately 50 to 75 gallons per minute. An operator can access these valves and equipment within approximately 45 minutes and initiate flow in approximately another 25 minutes.

Alternate Makeup Method No. 9 – Demineralized Water System to Spent Fuel Pools or Transfer Canals

34. This method provides water from the demineralized water system directly to the spent fuel pools or transfer canals. The west wall of the fuel handling building operating deck contains several service connections for demineralized water designed to connect hoses for use in filling or wash down activities on the fuel handling building operating deck. The demineralized water header is normally pressurized and, by connecting hoses located on the fuel handling building operating deck and opening one (1) header isolation valve, an operator can fill an accessible spent fuel pool or transfer canal at approximately 100 gallons per minute. An operator can access these valves and initiate flow in approximately 5 minutes.

TECHNICAL SUPPORT CENTER (TSC) EMERGENCY MANAGEMENT

35. The TSC is an Emergency Plan facility that is staffed upon declaration of an Alert or higher classification. The Site Emergency Coordinator ("SEC") is the lead position, in charge of coordinating TSC support of main control room response to plant emergencies. The TSC is staffed with representatives from key plant support groups including: operations, radiological control, security, offsite communications, and technical and accident assessment. The TSC is responsible for all on-site emergency response actions. The SEC is trained in the Severe Accident Mitigation Guidelines ("SAMGs"), which provide guidance for beyond design basis events that are not specifically covered in Emergency Operating Procedures. The SEC is in constant communication with the main control room, the operational support center ("OSC") (damage control missions are dispatched from this facility), and the emergency operations facility ("EOF") (coordinates off-site protective actions). The SEC receives important information from the main control room, such as notification of the loss of spent fuel cooling, and coordinates recovery missions with the OSC and control room. The TSC sets the priority for all site missions in response to accident conditions. A significant resource available to the SEC is the Accident Assessment Team, which consists of multi-disciplined, experienced engineers tasked with developing solutions to problems and conditions not specifically addressed in plant procedures.

36. The SAMGs are implemented when core temperatures exceed a pre-determined

value and efforts to restore cooling have been unsuccessful. The SAMGs are designed primarily for the Emergency Response Organization staff to fill the void between the EOPs and Emergency Plan regarding formalized guidance for severe accident situations. The primary goal of the SAMGs is to protect fission product barriers and mitigate any ongoing fission product releases, with a secondary goal to mitigate severe accident phenomena and return the plant to a controlled stable condition.

37. In evaluating an emergency, the TSC staff begins by monitoring a Diagnostic Flow Chart ("DFC") and the Severe Challenge Status Trees which provide guidance to diagnose challenges to fission product barriers. Predetermined values are used to direct implementation of specific SAMGs, which provide a number of solutions to potential challenges to a given barrier. The goals of the SAMGs are to 1) prevent core damage, 2) terminate the progress of core damage if it begins and retain the core in the reactor vessel, 3) maintain containment integrity as long as possible, and 4) minimize offsite releases. Work sheets that provide guidance on a number of mitigation options are included to aid in the development of an implementing strategy. The TSC then evaluates the current equipment capabilities and develops a course of action. The SEC also directs the Accident Assessment Team to develop strategies for the given situation. An analysis of the impacts of the strategies is conducted with the SEC and an strategy is selected and implemented.

CONCLUSIONS

38. There are numerous, diverse methods for providing cooling and makeup water to the Harris spent fuel pools following a loss of normal cooling.

39. Harris operators are trained and capable of performing the actions necessary to initiate one or more of these methods under emergency conditions, although the exact method or methods employed may depend on the specific plant conditions existing at the time. The necessary tools and equipment are available to perform the required actions.

40. Personnel assigned to the Harris TSC are familiar with the SAMGs and are trained to assist plant operators responding to emergency conditions outside of existing emergency response procedures.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2000.

Eric A. McCartney

Eric A. McCartney

Subscribed and sworn to before me
this 15 day of November 2000.

Darlene S. Yarborough

My Commission expires: 2-21-2005

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

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