EXHIBIT 3

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

AFFIDAVIT OF EVERETT L. REDMOND II, Ph.D.

COUNTY OF BURLINGTON	)	
	)	ss:
STATE OF NEW JERSEY	)	

I, Everett L. Redmond II, being duly sworn, do on oath state as follows:

EXPERIENCE AND QUALIFICATIONS

1. I am a nuclear engineer and have been with Holtec International ("Holtec"), since 1996. My business address is 555 Lincoln Drive West, Marlton, New Jersey, 08053.

2. Holtec is a diversified energy technology company working for the electric power industry both in the United States and in many countries around the world. Holtec performs the majority of its work for nuclear power plants. Holtec develops and markets turnkey equipment for the nuclear power industry. Holtec performs all of the design and engineering, obtains necessary governmental regulatory approvals, effectuates manufacturing, and performs on-site installation, testing, and commissioning into service of the products it sells. Holtec currently employs over 40 professional employees. A

large number of Holtec's employees hold graduate degrees from prestigious national and international universities, with a approximately 30 percent holding Ph.D.'s in science and engineering.

3. Holtec designs and markets both wet storage and dry storage systems for spent fuel storage. Holtec's expertise spans all aspects of spent fuel storage system development and supply including expertise in solid mechanics, heat transfer, nuclear physics, and nuclear components fabrication. One of Holtec's principal business areas is the design and installation of spent fuel storage racks for the expansion of wet storage at nuclear power plants. These projects included all of the design, analysis, and licensing reports required to obtain approval and implementation of the spent fuel storage rack capacity expansions. Holtec has a practically 100% market share in wet storage expansion. Holtec has completed turnkey projects for wet pool spent fuel storage capacity expansion in over 50 spent fuel pools in nuclear plants around the world.

4. I am one of Holtec's principal engineers responsible for performing nuclear criticality analyses for spent fuel storage systems. My expertise includes nuclear criticality analysis for both wet pool storage racks and dry cask storage systems for spent fuel storage. I am also Holtec's principal engineer for all radiation shielding analyses for spent fuel storage systems. I have over 10 years experience performing nuclear criticality analyses. I received my Ph.D. from the Massachusetts Institute of Technology ("M.I.T.") in the field of Nuclear Engineering. My doctoral dissertation focused in development of methodologies for performing nuclear shielding and criticality analyses. I also received both my Bachelor of Science and Master of Science degrees from M.I.T. in Nuclear

Engineering. I have supported the development of numerous nuclear criticality analyses for spent fuel storage systems designs to support license applications to the Nuclear Regulatory Commission ("NRC"). I have developed nuclear criticality analyses for both Pressurized Water Reactor and Boiling Water Reactor spent fuel pool reracking projects. I have also been the primary reviewer for the criticality analyses performed for numerous other spent fuel pool reracking projects, as well as Holtec's dry storage cask certificate of compliance applications. I have attached to this affidavit a copy of my resume with a list of my publications as Attachment A to this affidavit.

5. I make this affidavit to introduce my Report No. HI-992283 (Attachment B), explain the report's background, and summarize its principal conclusions. I also provide my review of the nuclear criticality analysis performed by the NRC Staff for this proceeding.

BACKGROUND AND OBJECTIVES OF REPORT

6. I performed the original nuclear criticality analysis for the Harris spent fuel pools C and D License Amendment Request. In addition to all the likely operating conditions attendant to the spent fuel storage racks, the original criticality analysis did analyze a boron dilution event, but did not explicitly analyze a fresh fuel assembly misplacement event.

7. My company, Holtec, had previously performed nuclear criticality analyses for the essentially identical spent fuel storage racks and identical fuel assembly characteristics to those to be used in Harris spent fuel pools C and D. These analyses specifically included a fresh fuel assembly misplacement event, assuming a fresh fuel

assembly with 5% enrichment in uranium-235, the same as the maximum permissible fresh fuel enrichment at Harris. The prior analysis had demonstrated that k-effective would remain well subcritical, below 0.95, if at least 400 ppm of soluble boron were included in the spent fuel pool water. This level of soluble boron in the pool water is far less than the level of 2000 ppm of soluble boron that is required to be maintained in the Harris spent fuel pools.

8. The NRC Staff submitted a request for additional information ("RAI") to CP&L requesting information regarding a fresh fuel assembly misplacement analysis. In response to this RAI, my company developed a response for CP&L which stated that the prior analysis, discussed above, had demonstrated that 400 ppm of soluble boron was sufficient to keep the spent fuel pool subcritical in the event of misplacement of a fresh fuel assembly. CP&L submitted this RAI response to the NRC on June 14, 1999. This RAI response is included as Attachment C to Exhibit 1, the Affidavit of R. Steven Edwards.

9. Following the admission of Contention 2, Basis 2 in this proceeding, concerning a fresh fuel assembly misplacement, I was requested by CP&L to perform a supplemental nuclear criticality analysis. The supplemental nuclear criticality analysis evaluates a fresh fuel assembly misplacement event specifically for the Harris Nuclear Plant and the spent fuel storage racks to be used in pools C and D.

10. Following consultation with CP&L, we agreed to perform the standard set of fuel assembly misplacement analyses required by the NRC Staff pursuant to their regulatory guidance and prior application approvals. Though the analysis had already

been performed for the same fuel and essentially the same storage racks for another plant, this analysis would be performed specifically for the Harris Nuclear Plant. This analysis evaluated the specific fuel assembly characteristics and spent fuel storage rack designs that are included in the Harris License Amendment Request to activate Harris spent fuel pools C and D.

11. The standard misplacement analysis required by the NRC Staff is a determination of k-effective for a single fresh fuel assembly of maximum permissible enrichment loaded into a spent fuel storage rack otherwise filled with spent fuel assemblies with the maximum permissible reactivity allowable under the proposed enrichment and burnup curve. The analysis is required to determine k-effective for the specific level of soluble boron required to be maintained in the plant's spent fuel pools (i.e., 2000 ppm for Harris), and to evaluate the minimum level of soluble boron necessary to maintain k-effective below 0.95. Both of these standard analyses were performed for the Harris spent fuel storage racks.

12. At CP&L's request, I also evaluated an additional scenario that exceeded the requirements of the NRC Staff. In addition to the standard misplacement analysis required by the NRC Staff, I also evaluated a hypothetical scenario where a single fresh fuel assembly of maximum permissible enrichment is loaded into a spent fuel pool containing absolutely no soluble boron (i.e., zero (0) ppm). This scenario was evaluated to demonstrate the conservatism present in the criticality control design for the spent fuel storage racks for Harris pools C and D.

APPROACH TO THE HARRIS MISPLACEMENT ANALYSIS

13. I performed the original nuclear criticality analysis ("Harris Base Criticality Analysis") for the spent fuel storage racks for Harris spent fuel pools C and D. This analysis is documented in the Holtec Licensing Report, HI-971760, entitled "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D." While Revision 3 of this report was filed in 1999, the substance of the Harris Base Criticality Analysis in the report has not changed since it was initially completed in 1998. This report is part of the License Amendment Request filed by CP&L with the NRC, which is included as Attachment A to Exhibit 1, the Affidavit of R. Steven Edwards. The analysis methodology, including the assumptions and modeling for the spent fuel storage rack design and the fuel assembly characteristics, used in the Harris Misplacement Analysis is identical to the methodology developed for, and used in, the Harris Base Criticality Analysis. The Harris Base Criticality Analysis is reference 6 of the Harris Misplacement Analysis.

14. The Harris Base Criticality Analysis used the CASMO-3 computer code to evaluate k-effective for the spent fuel storage racks. The Harris Misplacement Analysis used the MCNP-4A computer code to evaluate k-effective. These results were compared against the CASMO-3 results in the Harris Base Criticality Analysis to demonstrate that the calculations are bounding. Both CASMO-3 and MCNP-4A are standard industry computer codes used for performing nuclear criticality analyses. The CASMO-3 and MCNP-4A computer codes used by Holtec for criticality analyses are verified and validated pursuant to Holtec's quality assurance program.

15. The Harris Misplacement Analysis analyzes the specific spent fuel storage racks to be used in Harris spent fuel pools C and D.

16. The Harris Misplacement Analysis analyzes the specific fuel characteristics of the maximum reactivity spent fuel that could be stored in the spent fuel storage racks to be used in Harris spent fuel pools C and D. The maximum reactivity fuel assembly type is the Westinghouse 15x15 PWR fuel assembly. The fresh fuel assembly with the maximum permissible reactivity at Harris is fuel enriched to 5% (by weight) in uranium-235. The Harris Misplacement Analysis assumes that a single fresh fuel assembly of the maximum permissible enrichment is loaded into a storage rack otherwise containing spent fuel of the maximum reactivity permissible under the enrichment and burnup curve for the Harris pools C and D spent fuel storage racks. The Harris Misplacement Analysis considers an infinite array of such maximum reactivity fuel. This analysis of k-infinity provides an additional degree of conservatism because it results in a higher neutron multiplication factor ("k") than the analysis of k-effective, because k-infinity neglects the reduction in reactivity due to neutron leakage from a finite array of fuel assemblies.

17. The Harris Misplacement Analysis takes into account all the same uncertainties used in the Harris Base Criticality Analysis, including calculational uncertainties, manufacturing tolerances, and burnup uncertainties. All uncertainties were statistically combined in order to demonstrate that the calculated k-effective is known with a 95% probability at a 95% confidence level, in accordance with NRC guidance.

18. The Harris Misplacement Analysis was reviewed and approved by both Holtec's internal review process and by CP&L's review process. The Harris Misplacement Analysis was verified and validated through the Holtec quality assurance process, which included an independent review and approval of the analysis by another competent nuclear criticality analyst. The Harris Misplacement Analysis was also reviewed by CP&L through the Owner's Review process. The CP&L review resulted in approval of the analysis with no adverse comments.

CONCLUSIONS OF THE HARRIS MISPLACEMENT ANALYSIS

19. The analysis methods, assumptions, and the specific misplacement scenarios evaluated in the Harris Misplacement Analysis are consistent with those required by the NRC Staff. The Harris Misplacement Analysis is consistent with both NRC Staff guidance documents and with Staff practice with regard to previous Holtec spent fuel storage rack license amendment applications that have been approved by the Staff.

20. The Harris Misplacement Analysis demonstrates that the spent fuel storage racks in Harris pools C and D will remain subcritical even following the misplacement of a fresh fuel assembly with the maximum permissible enrichment at Harris into the spent fuel storage racks with no soluble boron.

21. The analysis demonstrates that the spent fuel storage racks, with the required 2000 ppm of soluble boron in the spent fuel pool water, will remain subcritical following the misplacement event, with a k-effective of 0.7783.

22. The analysis confirms that the spent fuel storage racks will remain subcritical, with a k-effective of less than 0.95, following the misplacement event with an assumed 400 ppm of soluble boron in the pool water. This is far below the 2000 ppm of soluble boron required to be maintained in the spent fuel pool water by Harris operating procedures. With only 400 ppm of soluble boron assumed in the pool water, k-effective was demonstrated to be 0.9352. This result confirms the response made by CP&L in its June 14, 1999 RAI response to the NRC (Attachment C to Exhibit 1).

23. The additional analysis performed for the Harris Misplacement Analysis demonstrates that the spent fuel storage racks will remain subcritical following a fresh fuel assembly misplacement event even if <u>no</u> soluble boron (<u>i.e.</u>, zero (0) ppm) is present in the spent fuel pool water, with a k-effective of 0.9932. This analysis simulates a complete (i.e., 100%) dilution event, in which <u>all</u> of the soluble boron is hypothetically assumed to be removed from the spent fuel pool water concurrent with the fuel assembly misplacement event. While not required to be evaluated under the NRC Staff's double contingency principle, this analysis demonstrates that the issue of soluble boron dilution is moot with respect to a fuel assembly misplacement event in the spent fuel storage racks for Harris pools C and D.

NRC Staff's Criticality Analysis

24. In November, 1999, the NRC Staff performed an independent nuclear criticality analysis of multiple fuel assembly misplacements for this proceeding. The Staff's criticality analysis was performed by Tony P. Ulses, a nuclear engineer in the NRC Staff's Reactor Systems Branch. This analysis is documented in the NRC Staff's

November 5, 1999 memorandum and report, which is included as Attachment C to this affidavit. The Staff's analysis assumes the concurrent misplacement of an infinite number of fresh fuel assemblies of the maximum permissible reactivity at Harris. The Staff's analysis utilized boundary conditions that are reflective in the x, y, and z directions, which models an infinite array of fresh fuel assemblies. The analysis includes the effects of the soluble boron required to be present in the spent fuel pools pursuant to plant operating procedures. This analysis is not required under the double contingency principle in the Staff's regulatory guidance, since even <u>two</u> fresh fuel assembly misplacements are two independent, unlikely, concurrent events. The NRC Staff's analysis of an <u>infinite</u> number of concurrently misplaced fresh fuel assemblies of the maximum possible reactivity is far beyond what is considered a credible event for analysis purposes.

25. I have reviewed the NRC Staff's November 5, 1999 memorandum and report on its misplacement criticality analysis. I am familiar with the analysis methodology, assumptions, and computer codes used in the Staff's analysis. Based on my review, I have determined that the Staff modeled the most reactive fresh fuel assemblies permissible at Harris and the spent fuel storage racks to be used for those assemblies in Harris spent fuel pools C and D. The Staff's analysis concluded that the spend fuel storage racks will remain subcritical, with a calculated k-effective of 0.98. The Staff's analysis did not consider manufacturing tolerances, but assumed that the k-effective bias from manufacturing tolerances is not larger than 1%. I am familiar with the manufacturing tolerances applicable to these spent fuel storage racks, and I confirm that

the bias from these manufacturing tolerances is 0.0048, less than 1%.

26. Based on my review and my experience performing nuclear criticality analyses, I confirm the results of the nuclear criticality analysis performed by the NRC Staff. The results of the analysis are consistent with my expectations based on my knowledge of the spent fuel storage rack designs, fresh fuel assembly characteristics, analytical methods, and calculations.

27. The NRC Staff's analysis demonstrates that the spent fuel storage racks for Harris spent fuel pools C and D will remain subcritical, even if <u>every</u> location in the spent fuel storage rack is assumed to be concurrently loaded with a misplaced fresh fuel assembly of the maximum possible reactivity at the Harris Nuclear Plant. While this analysis is not required under the Staff's double contingency principle, the NRC Staff's criticality analysis of an <u>infinite</u> number of fresh fuel assembly misplacements demonstrates that the issue of multiple fuel assembly misplacements is moot with respect to the spent fuel storage racks for Harris pools C and D.

I declare under penalty of perjury that the foregoing statements and my statements in the attached report are true and correct.

Everett L. Redmond II

Everett L. Redmond II
December 29, 1999

Subscribed and sworn to before me this

29th day of December, 1999

My commission expires 5-26-02

Craig Smith

CRAIG SMITH
NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES MAY 26, 2002

EVERETT L. REDMOND II, Ph.D.

PRINCIPAL ENGINEER
HOLTEC INTERNATIONAL

EDUCATION

Massachusetts Institute of Technology
Ph.D. in Nuclear Engineering and a Minor in Biology (1997)

Massachusetts Institute of Technology
M.S. in Nuclear Engineering (1990)

Massachusetts Institute of Technology
B.S. in Nuclear Engineering (1990)

PROFESSIONAL EXPERIENCE

HOLTEC INTERNATIONAL
Marlton, New Jersey
1995 – Present — Principal Engineer

HOLTEC INTERNATIONAL
Palm Harbor, Florida
August 1994 – Spring 1995 — Criticality and Shielding Consultant

LOS ALAMOS NATIONAL LABORATORY
Los Alamos, New Mexico
Summers 1993 and 1994 — Graduate Research Assistant

RAYTHEON
Sudbury, Massachusetts
Spring 1993 — Shielding Consultant

NORTHEAST UTILITIES COMPANY
Hartford, Connecticut
Summer 1992 — Engineer

IDAHO NATIONAL ENGINEERING LABORATORY
Idaho Falls, Idaho
Summers 1987, 1988, 1990 — Engineer and Co-op Student
June 1989 - January 1990

PROFESSIONAL SOCIETY MEMBERSHIPS/ACTIVITIES

Member American Nuclear Society (1986-Present)

DRY AND WET SPENT FUEL STORAGE TECHNOLOGY

- Developed Holtec's shielding analysis methods for dry cask storage licensing.
- Developed Holtec's shielding analysis methods and models for performing site boundary dose calculations for an ISFSI.
- Created all computer models of HI-STAR 100, HI-STORM 100, 100-ton and 125-ton HI-TRACs used in the shielding analysis reported in the HI-STAR SAR and HI-STAR and HI-STORM TSARs under Dockets 71-9261, 72-1008, and 72-1014
- Author of Shielding Evaluation Chapters in the HI-STAR SAR and HI-STAR and HI-STORM TSARs under Dockets 71-9261, 72-1008, and 72-1014
- Primary reviewer for Criticality Evaluation Chapters in the HI-STAR SAR and HI-STAR and HI-STORM TSARs under Dockets 71-9261, 72-1008, and 72-1014
- Performed criticality analysis for both PWR and BWR spent fuel pool reracking.
- Served as primary reviewer for numerous criticality analyses for spent fuel pool reracking.

PUBLICATIONS

1. E.L. Redmond II, " Methodology for Calculating Dose Rates from Storage Cask Arrays Using MCNP," *Trans. Am. Nucl. Soc.*, 77, 332, (1997)

2. E.L. Redmond II, "Multigroup Cross Section Generation Via Monte Carlo Methods," Doctoral Thesis, Massachusetts Institute of Technology (1997).

3. R. Zamenhof, E. Redmond II, G. Solares, D. Katz, K. Riley, S. Kiger, and O. Harling, "Monte-Carlo-Based Treatment Planning for Boron Neutron Capture Therapy Using Custom Designed Models Automatically Generated From CT Data," Int. J. Radiation Oncology Biol. Phys., 35 383-397 (1996).

4. O.K. Harling, R.D. Rogus, E.L. Redmond II, K.A. Roberts, D.J. Moulin and C.S. Yam, "Phantoms for Neutron Capture Therapy Dosimetry," presented at Sixth International Symposium on Neutron Capture Therapy for Cancer, Kobe, Japan, October 31 - November 4, 1994.

5. J.C. Wagner, E.L. Redmond II, S.P. Palmtag, J.S. Hendricks, "MCNP: Multigroup/Adjoint Capabilities," LA-12704, Los Alamos National Laboratory (1994).

6. E.L. Redmond II, J.C. Yanch, and O.K. Harling, "Monte Carlo Simulation of the MIT Research Reactor," *Nuclear Technology, 106*, 1, April 1994.

7. E.L. Redmond II and J.M. Ryskamp, "Monte Carlo Methods, Models, and Applications for the Advanced Neutron Source," *Nuclear Technology, 95*, 272, (1991).

8. R.C. Thayer, E.L. Redmond II, and J.M. Ryskamp, "A Monte Carlo Method to Evaluate Heterogeneous Effects in Plate-Fueled Reactors," *Trans. Am. Nucl. Soc., 63*, 445, (1991).

9. J.M. Ryskamp, E.L. Redmond II and C.D. Fletcher, "Reactivity Studies on the Advanced Neutron Source Preconceptual Reactor Design," *Proc. Topl. Mtg. Safety of Non-Commercial Reactors*, Boise, ID, October 1-4, 1990, Vol. I, p. 337 (1990).

10. E.L. Redmond II and J.M.Ryskamp, "Monte Carlo Methods, Models, and Applications for the Advanced Neutron Source," *Trans. Am. Nucl. Soc.*, *61*, 377 (1990).

11. E.L. Redmond II, "Monte Carlo Methods, Models, and Applications for the Advanced Neutron Source," Masters Thesis, Massachusetts Institute of Technology (1990).

12. E.L. Redmond II and J.M. Ryskamp, "Design Studies on Split Core Models with Involute Fuel for the Advanced Neutron Source," NRRT-N-88-034, Idaho National Engineering Laboratory (1988).

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See Notebook
Containing Proprietary
Information

EVALUATION OF FRESH FUEL ASSEMBLY MISLOAD IN HARRIS POOLS C AND D

FOR

CAROLINA POWER AND LIGHT

Holtec Report No: HI-992283

Holtec Project No: 90864

Report Category: A

Report Class : SAFETY RELATED

November 5, 1999

MEMORANDUM TO: Ralph Caruso, Chief
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Tony P. Ulses, Nuclear Engineer /s/
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: COMPLETION OF CRITICALITY ASSESSMENT OF MISLOADING ERROR IN HARRIS C AND D SPENT FUEL POOL

I have completed the analysis evaluating the potential for criticality from a misloading error if Shearon Harris begins to use high density storage racks in the currently inactive C and D spent fuel pools. The analysis discussed in the enclosed report assumes a worst case misloading error in which the entire rack is misloaded with fresh 5 w/o enriched Westinghouse 15x15 fuel which has been previously determined to be the most reactive PWR fuel type which could be loaded into the Harris pools. This analysis demonstrates that the multiplication factor will remain less than one (i.e. subcritical) for this postulated worst case scenario. The calculated eigenvalues are taken at upper 95/95 level and a manufacturing uncertainty of 1 percent has been added to the predicted value.

Enclosures:
As stated

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Evaluation of Postulated Worst Case Misloading Error for Harris C and D Spent Fuel Pools

Tony P. Ulses
November 2, 1999

1 Introduction

Carolina Power and Light (CP&L), the operator of the Shearon Harris nuclear power plant, requested a license amendment to activate the two unused spent fuel pools at the Harris site. The proposal is to use a "high density" storage configuration which requires the use of burnup credit racks. In the context of this report burnup credit racks refer to storage racks which require that the fuel has reached a pre-specified minimum burnup before it can be safely stored. The need for this burnup requirement is dictated by the fact that the inter-assembly spacing is reduced to achieve the desired "high density" configuration. Whenever one relies on a physical process such as burnup one needs to assess the impact of an assembly being inserted into the rack that has not reached the minimum acceptable burnup. Therefore, criticality analyses have been performed to assess the effect of an assembly misloading error in the Harris "C" or "D" spent fuel pool. In this analyses it was assumed that the entire rack was misloaded with UO_2 fuel enriched to 5 w/o U^{235} which is the highest enrichment allowed at commercial power plant's in the US. This would be the worst possible configuration.

2 Definition of Problem

In this analyses we will assess the impact of a worst case misloading accident by predicting the multiplication factor of the system. To this end, we will perform three base analyses and one sensitivity calculation. Two of the base analyses are intended to assess the staff's criticality calculations against the licensee calculations and the final analyses will assess the worst case misloading accident. The two comparative calculations are important because they will allow an assessment of the licensee method's and will serve to strengthen the staff's position with respect to these methods. A brief description of the problems will follow:

<u>Typical Parameters</u>

Fuel type:	Westinghouse 15x15 Assembly Enriched to 5 w/o U^{235}
Rack type:	Holtec High Density
Boundary Conditions:	Reflective in x, y, and z
# of Histories:	1000 groups of 3000 particles for a total of 3 million histories

<u>Problem 1</u>

This problem is extracted from reference 1. The rack should be assumed to be loaded with fresh fuel without soluble boron. All dimensions should be nominal.

<u>Problem 2</u>

This problem is the licensing basis for the storage racks. The rack should be loaded with fuel burned to 41.7 Mw_d/KgU. The depletion is to be performed assuming three cycles of operation with an average boron concentration of 900 ppm, a specific power of 42 kW/KgU, nominal fuel and clad temperature and slightly higher than expected moderator temperature. The criticality analyses should assume no soluble boron is present and credit will be taken for actinides and fission products. All dimensions should be nominal.

Problem 3

This problem assesses the effect of the worst case misloading accident. The rack should be loaded with fresh fuel and one should assume that the soluble boron is present. All dimensions should be nominal.

3 Description of Methods

The SCALE (ref. 2) system was chosen for both the criticality analyses and the burnup calculations. The SCALE system has been extensively assessed and validated for these types of calculations (refs. 3 - 5). The SAS2H sequence was used for the depletion calculations and the CSAS6 sequence was used for the criticality calculations. Both of these sequences use BONAMI and NITAWL-II to process cross sections into a problem specific AMPX working library. SAS2H uses XSDRN and ORIGEN to deplete the fuel and CSAS6 uses KENO-VI for criticality calculations. Both the 44 group and the 238 group ENDF/B-V based AMPX libraries were used in the criticality analyses and the 44 group AMPX library was used for depletion.

4 Presentation and Discussion of Results

The results for problems 1 and 2 are presented in table 1. For comparative purposes, we have included the results from the licensee's contractor (ref. 1). This comparison reveals that the licensee method seems to predict slightly higher mulitplication factors (as much as 2% overall). However, given the differences in the methods the staff considers this to be excellent agreement and this gives us a great deal of confidence in the methods being used by both the staff and the licensee.

Table 1 Comparison of Results for Problem 1 and Problem 2

	CASMO	MCNP	SCALE[1]
Problem 1	1.2076	1.2056	1.19378
Problem 2	0.9126	N/A	0.8940

[1]The SCALE results are the staff calculation.

The multiplication factor predicted for problem 3 is 0.978 at the upper 95/95 interval using the 44 group library and 0.979 using the 238 group library. The 238 group library was also used for this problem to ensure that collapsing spectrum used to generate the 44 group library from the 238 group library did not introduce any significant bias into the results. This demonstrates that even assuming the worst case misloading error (i.e. misloading an entire rack with fresh fuel) the rack will remain subcritical when one considers the soluble boron which will be present in the pool.

In order to assess the adequacy of multiplication factors predicted using Monte Carlo methods it is prudent to consider, in addition to the number of histories tracked, how well the spatial and energy domains of the problem were sampled. To this end, we have attached the spectrum output for the global unit from KENO-VI in Appendix A and prepared several spectral plots. The information from the major edit indicates that all of the parts of the problem have been sampled. Note that the flux for region 1 in the global unit is zero because region 1 represents the hole containing the fuel which was inserted into the global unit. The flux should be zero in the global unit for this region.

The spectral plots are presented as Figures 1 and 2. The error bars represent one standard deviation and were extracted from the major edit (see Appendix A). From these plots we can ascertain that there are no unexpected trends in the results. For example, figure 1 shows a characteristic light water moderated reactor spectrum, but the thermal peak is smaller than it would be in the reactor. This reduction is caused by the additional absorption in the rack poison. Furthermore, we can see that we had complete coverage of the energy domain and that the sampling was significant enough to reduce the standard deviation to acceptable values.

5 Conclusions

Analyses have been performed to assess the effect of the worst case misloading scenario in the Harris "C" and "D" spent fuel pool. This analysis demonstrates that the maximum possible multiplication factor in the "C" and "D" spent fuel pools is 0.98 assuming that one credits the soluble boron present in the pool coolant. It should be noted that this analysis does not consider manufacturing tolerances, but the multiplication factor bias from manufacturing uncertainties is typically not larger than 1%. The staff has also been able to confirm that the methods used by the licensee contractor yield results that are consistent with the staff's results.

6 References

1. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998. (Holtec International Proprietary)

2. "SCALE 4.3, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Oak Ridge National Laboratory, 1995.

3. M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.

4. O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

5. W. C. Jordan, et. al., "Validation of KENOV.a, Comparison with Critical Experiments" ORNL/CSD/TM-238, Oak Ridge National Laboratory, 1986.

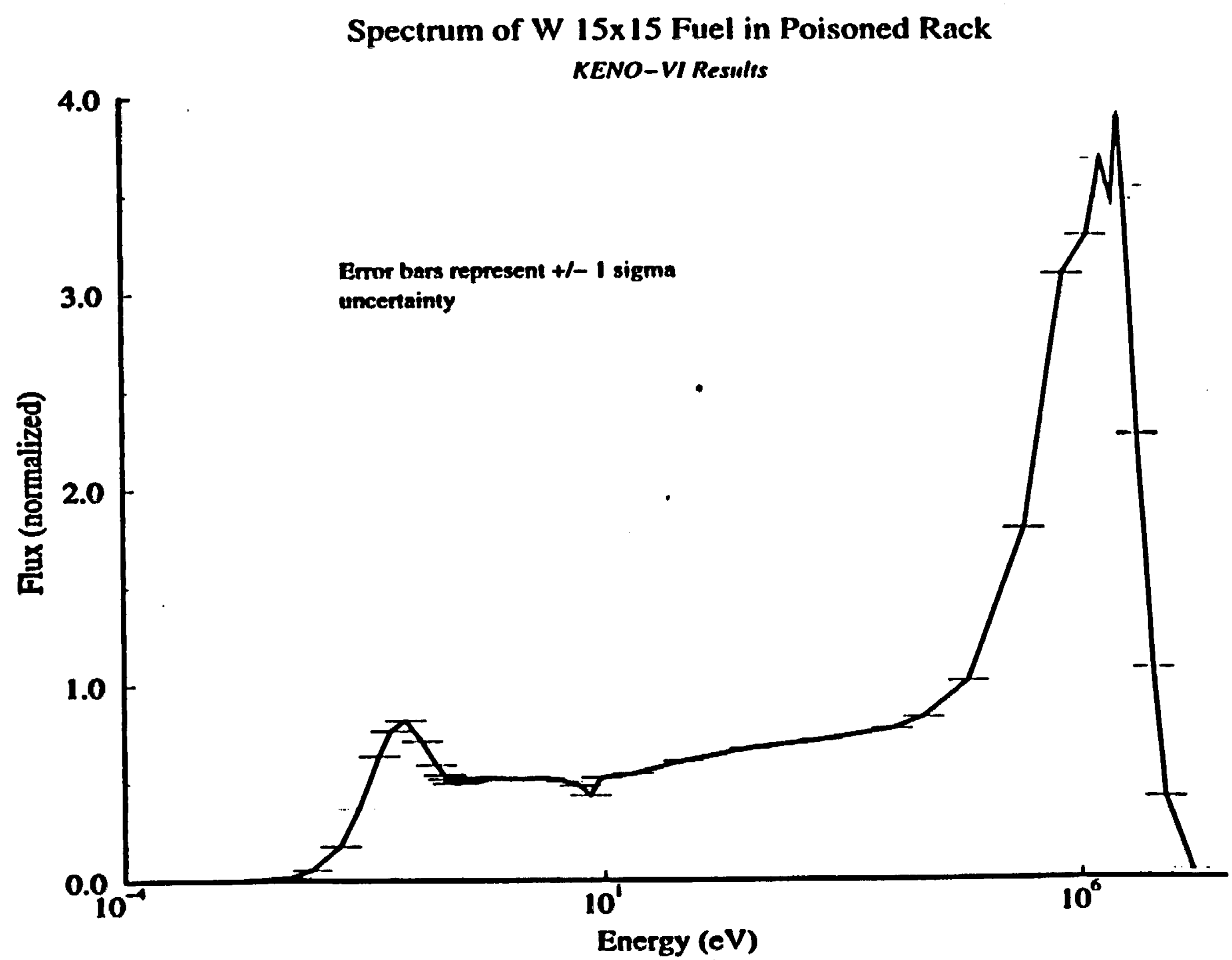

Figure 1 KENO Predicted Spectrum for W 15x15 Fuel Assembly

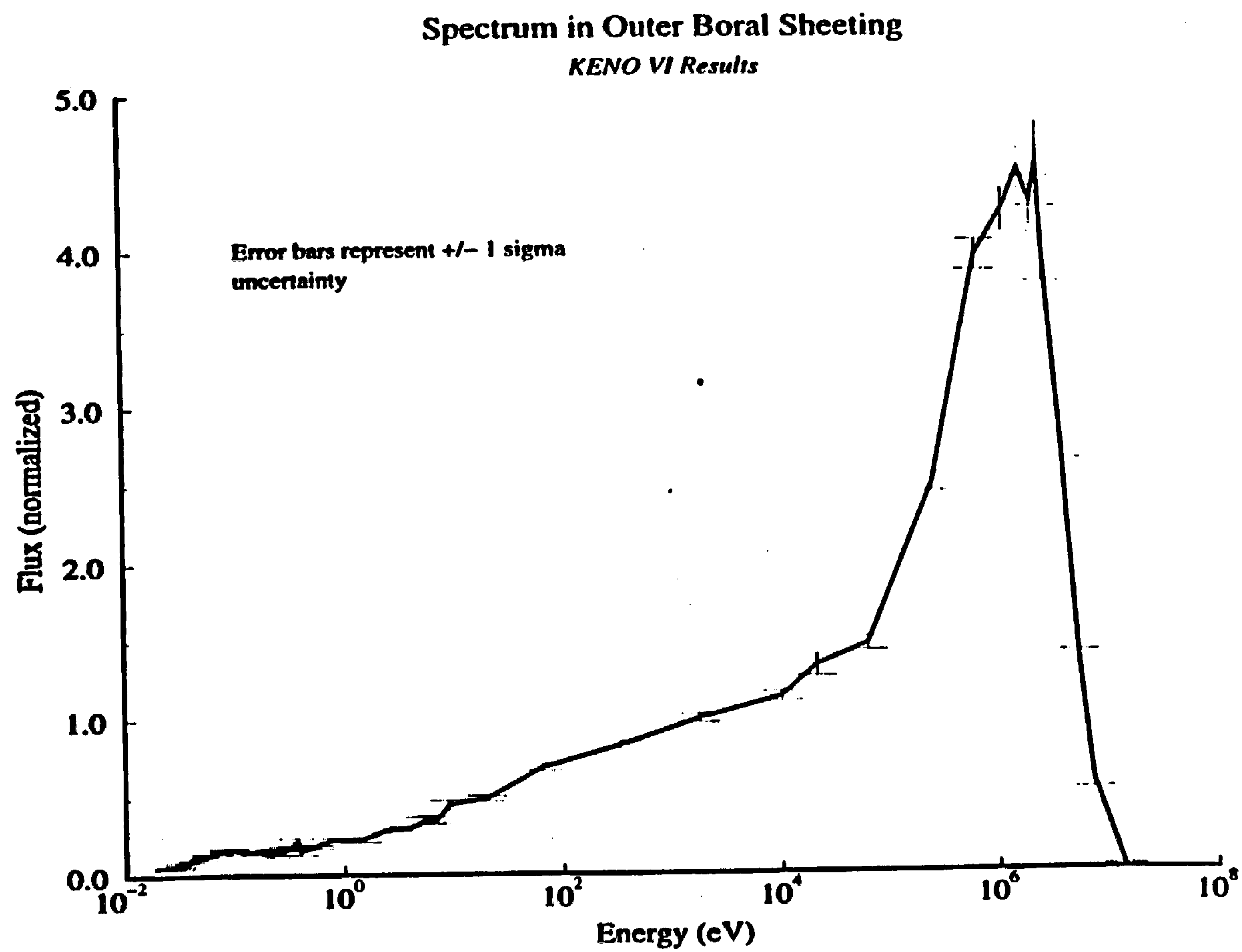

Figure 2 KENO Predicted Spectrum in Outer Boral Sheeting

Appendix A

Excerpt from KENO-VI Major Edit

1 keno-vi input for stor[illegible] cell calc. for holtec rack w/ 15x15 w

0fluxes for global unit

0group	region 1 flux	region 1 percent deviation	region 2 flux	region 2 percent deviation	region 3 flux	region 3 percent deviation	region 4 flux	region 4 percent deviation	region 5 flux	region 5 percent deviation	region 6 flux	region 6 percent deviation
1	0.000E+00	0.00	1.376E-04	5.36	8.973E-06	18.11	1.932E-05	19.59	1.823E-05	18.58	2.150E-05	12.41
2	0.000E+00	0.00	4.190E-04	3.46	5.856E-05	8.68	5.653E-05	8.31	4.404E-05	9.44	4.438E-05	8.41
3	0.000E+00	0.00	1.267E-03	1.92	1.656E-04	5.23	1.544E-04	5.92	1.281E-04	5.35	1.475E-04	4.97
4	0.000E+00	0.00	4.204E-03	1.14	5.437E-04	3.46	5.072E-04	3.55	4.983E-04	3.51	4.957E-04	2.91
5	0.000E+00	0.00	2.834E-03	1.37	3.175E-04	3.97	3.393E-04	3.74	3.345E-04	4.01	3.336E-04	3.86
6	0.000E+00	0.00	8.974E-04	2.41	9.913E-05	5.97	1.171E-04	7.88	1.041E-04	6.83	1.040E-04	6.43
7	0.000E+00	0.00	3.574E-03	1.33	4.377E-04	3.59	4.251E-04	3.72	3.972E-04	4.34	4.402E-04	3.67
8	0.000E+00	0.00	4.386E-03	1.18	5.304E-04	3.18	5.120E-04	3.19	4.895E-04	3.44	5.272E-04	3.33
9	0.000E+00	0.00	6.307E-03	1.08	7.926E-04	3.15	7.232E-04	3.15	6.767E-04	3.19	7.520E-04	2.96
10	0.000E+00	0.00	1.103E-02	0.78	1.355E-03	2.44	1.291E-03	2.43	1.246E-03	2.54	1.271E-03	2.40
11	0.000E+00	0.00	1.178E-02	0.74	1.464E-03	2.26	1.336E-03	2.36	1.340E-03	2.54	1.391E-03	2.39
12	0.000E+00	0.00	7.178E-03	0.92	8.611E-04	3.00	8.099E-04	2.97	7.276E-04	3.03	7.950E-04	2.93
13	0.000E+00	0.00	1.595E-03	1.71	2.171E-04	5.22	1.834E-04	5.38	1.810E-04	5.70	1.772E-04	5.20
14	0.000E+00	0.00	7.130E-03	0.92	8.294E-04	2.75	7.465E-04	2.97	7.295E-04	3.31	8.029E-04	2.87
15	0.000E+00	0.00	6.261E-03	0.92	7.122E-04	2.72	6.722E-04	2.81	6.210E-04	2.98	6.581E-04	2.97
16	0.000E+00	0.00	5.505E-03	0.92	5.951E-04	2.66	5.580E-04	2.90	5.222E-04	2.90	5.567E-04	2.73
17	0.000E+00	0.00	3.273E-03	1.15	3.484E-04	3.02	3.119E-04	3.32	2.897E-04	3.31	3.040E-04	3.06
18	0.000E+00	0.00	2.444E-03	1.36	2.262E-04	3.42	2.102E-04	3.45	2.065E-04	3.42	2.197E-04	3.25
19	0.000E+00	0.00	4.374E-04	2.80	3.954E-05	7.46	3.452E-05	6.89	3.002E-05	7.71	3.751E-05	8.08
20	0.000E+00	0.00	5.568E-04	2.75	4.471E-05	6.49	4.308E-05	6.56	3.613E-05	6.99	4.229E-05	6.87
21	0.000E+00	0.00	4.168E-04	2.97	3.427E-05	6.93	2.959E-05	7.91	3.034E-05	7.48	3.174E-05	7.49
22	0.000E+00	0.00	7.767E-04	2.22	5.679E-05	5.29	6.221E-05	5.63	5.160E-05	5.67	5.866E-05	5.56
23	0.000E+00	0.00	8.810E-04	2.08	6.311E-05	4.74	6.017E-05	4.94	5.510E-05	4.97	6.113E-05	4.85
24	0.000E+00	0.00	9.433E-04	1.98	5.403E-05	5.21	5.279E-05	5.27	5.165E-05	5.02	5.716E-05	4.83
25	0.000E+00	0.00	7.081E-04	2.24	4.488E-05	5.09	3.932E-05	5.31	3.755E-05	5.45	3.815E-05	5.11
26	0.000E+00	0.00	6.778E-04	2.21	3.444E-05	5.51	3.357E-05	5.09	2.928E-05	5.60	3.339E-05	5.59
27	0.000E+00	0.00	8.796E-05	4.92	4.091E-06	13.38	5.436E-06	12.26	4.366E-06	15.39	4.106E-06	14.27
28	0.000E+00	0.00	9.516E-05	5.12	6.096E-06	12.92	4.348E-06	14.18	3.879E-06	14.43	4.893E-06	13.21
29	0.000E+00	0.00	1.080E-04	4.50	5.983E-06	13.01	5.313E-06	15.04	5.081E-06	12.86	6.356E-06	11.50
30	0.000E+00	0.00	2.454E-04	3.50	1.201E-05	8.66	1.019E-05	11.42	1.107E-05	8.96	1.019E-05	8.98
31	0.000E+00	0.00	1.288E-04	3.78	5.914E-06	11.15	6.818E-06	13.22	5.718E-06	12.02	5.699E-06	12.93
32	0.000E+00	0.00	1.513E-04	3.91	6.783E-06	10.57	6.677E-06	10.90	6.587E-06	11.27	7.080E-06	9.90
33	0.000E+00	0.00	1.739E-04	3.52	6.496E-06	9.67	6.806E-06	9.95	7.721E-06	10.56	6.509E-06	10.88
34	0.000E+00	0.00	4.281E-04	2.47	1.835E-05	6.12	1.563E-05	6.45	1.648E-05	6.63	1.735E-05	6.08
35	0.000E+00	0.00	6.916E-04	1.90	2.472E-05	5.58	2.395E-05	5.19	2.208E-05	5.36	2.412E-05	5.32
36	0.000E+00	0.00	6.888E-04	1.78	2.515E-05	4.84	2.490E-05	5.24	2.035E-05	6.30	2.428E-05	4.73
37	0.000E+00	0.00	5.795E-04	1.93	1.896E-05	5.63	1.813E-05	5.20	1.732E-05	5.52	1.871E-05	5.04
38	0.000E+00	0.00	3.240E-04	2.18	1.036E-05	7.29	8.607E-06	7.85	1.001E-05	7.77	9.824E-06	8.23
39	0.000E+00	0.00	3.261E-04	2.37	9.701E-06	8.07	8.411E-06	7.75	7.653E-06	7.96	9.549E-06	7.42
40	0.000E+00	0.00	1.468E-04	3.28	3.917E-06	10.32	4.053E-06	10.54	3.024E-06	12.74	3.141E-06	11.64
41	0.000E+00	0.00	3.566E-04	2.29	1.058E-05	6.57	9.020E-06	7.10	8.858E-06	7.00	9.430E-06	6.96
42	0.000E+00	0.00	3.604E-05	5.88	1.009E-06	27.70	8.304E-07	19.11	8.564E-07	25.72	8.251E-07	21.12
43	0.000E+00	0.00	3.968E-05	5.42	9.087E-07	18.02	1.018E-06	16.41	8.949E-07	30.82	6.629E-07	20.54
44	0.000E+00	0.00	6.744E-06	11.51	2.102E-07	37.98	1.685E-07	40.02	1.729E-08	70.77	2.139E-07	37.00