UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
NORTHEAST NUCLEAR ENERGY CO.	)	Docket No. 50-336-OLA
	)	(Spent Fuel Pool Design)
(Millstone Nuclear Power Station,	)	
Unit No. 2)	)	

AFFIDAVIT OF DR. STANLEY E. TURNER

I, Dr. Stanley E. Turner, being duly sworn, hereby state as follows:

1. I am employed by HOLTEC INTERNATIONAL (HOLTEC) as Chief Scientist, responsible for all nuclear analyses. This includes criticality safety analyses of new and spent fuel storage racks, radiological impact evaluations, and fuel management studies. I am a member of the ANS Standards Committee 8.17 on criticality safety of fuel in storage and, to date, have performed the criticality safety analysis for approximately 25 fuel storage pools in this country and abroad. I am also responsible for measurements of Boraflex degradation (by so-called Blackness tests and by examination of surveillance coupons removed from storage pools), and have made such measurements for numerous plants in this country and abroad.

A copy of my qualifications showing my complete professional experience is included with this Affidavit.

2. My business address and phone number are:

HOLTEC INTERNATIONAL
230 Normandy Circle
Palm Habor, FL 34683
(813) 787-4625

3. I was specifically responsible for performing for Northeast Utilities Service Company (NUSCo) the criticality analysis that supported the Millstone Unit No. 2 (Millstone 2) License Amendment 158. I also performed Blackness tests to determine the state of Boraflex degradation in the Millstone 2 spent fuel storage racks.

OVERVIEW

4. I have reviewed the two affidavits ("declarations") provided to date in this proceeding by Dr. Michio Kaku on behalf of Cooperative Citizens Monitoring Network. These affidavits were dated August 1992 and March 1993. In addition, I have reviewed the four subissues as stated by the Atomic Safety and Licensing Board (ASLB) in admitting Contention 1 in this proceeding.

5. Based upon my review of Dr. Kaku's materials, a review of the ASLB issues, and my work on the Amendment 158 criticality analysis for NUSCo, I conclude that no facts have been

presented to indicate that a genuine dispute exists in this proceeding.

6. First, Dr. Kaku does not appear to me to have any knowledge of the Boraflex conditions at Millstone 2, and does not even acknowledge the extensive Blackness testing campaigns carried out at Millstone 2 to characterize the present condition of the Boraflex panels. The Amendment 158 criticality analysis, as described further below, expressly incorporated assumptions carefully selected to very conservatively bound the actual condition of the Boraflex. Dr. Kaku provides no factual basis on which to question these assumptions.

7. In general, it also appears that Dr. Kaku has not read the criticality analysis report that supports Amendment 158. In particular, he apparently was not aware of the benchmarking of the analysis performed against critical experiments with thin absorbers, simulating as closely as possible the storage racks at Millstone 2. For example, in his March 1993 affidavit, at ¶ 12, Dr. Kaku states that "no benchmarking of the spent fuel pond with Boraflex boxes was ever done, which calls into question the reliability of the KENO codes." As I will demonstrate below, this statement is based on a factual premise that is simply not correct.

8. Many of Dr. Kaku's comments also reveal a confusion between diffusion theory and transport theory calculations (see for example, page 6 of Dr. Kaku's March 1993 affidavit); only the latter type analysis was utilized for Amendment 158. Nor does Dr. Kaku seem aware of the independent verification analyses performed by an entirely different method of analysis (as will be discussed below). Apparently Dr. Kaku had not reviewed the discovery materials provided by Northeast Utilities which show that independent verification indeed was performed.

9. Dr. Kaku's discussion of Monte Carlo reveals a lack of understanding of the technique and, in many cases, he makes statements that can only be characterized as unreasonable and without any substantive basis. See, for example, the March 1993 affidavit, page 17 item (d), and pages 13-15. None of his allegations present any information that would seriously challenge the validity of the Amendment 158 criticality safety analysis of the Millstone 2 racks.

CRITICALITY ANALYSIS

10. As described in the separate Affidavit of John R. Guerci, the Amendment 158 criticality analysis utilizes a three dimensional NITAWL-KENO-5a model with the 27-group Standardized Computer Analysis for Licensing Evaluation (SCALE) cross-section set developed for the Nuclear Regulatory Commission (NRC) by the Oak Ridge National Laboratory (ORNL). NITAWL-KENO is a

standard and accepted method of criticality analysis that has been used for many storage rack analyses, and for all racks (approximately 25) designed by HOLTEC or HOLTEC personnel.

11. NITAWL-KENO-5a is part of the SCALE package developed by ORNL. The original KENO code was developed in the late 1960s at ORNL and has been improved since that time. The KENO code utilizes a Monte Carlo statistical simulation of the Boltzman neutron transport theory.

12. Dr Kaku in his August 1992 affidavit seems to argue almost entirely with the use of diffusion theory as a means to perform a criticality analysis. However, the Amendment 158 analysis, as stated above, is premised on transport rather than diffusion theory. Transport theory provides a much more accurate description of neutron distribution than does diffusion theory. The Monte Carlo simulation has long been considered to be the closest analytical technique to an actual experiment for criticality analyses. Dr. Kaku's comments on diffusion theory analysis are completely irrelevant.

13. The criticality analysis for Amendment 158 incorporates many inherent conservatisms as described by Mr. Guerci in his Affidavit.

14. With respect to Boraflex degradation (i.e., gap formation), the following additional conservative assumptions were made:

- a 5.65-inch gap (4%) in <u>all</u> Boraflex panels, with a random gap distribution;
- a 4% shrinkage in width in each panel.

Each of these assumptions significantly exceeds what has been observed at Millstone 2 through the Blackness testing campaigns described by Mr. Betancourt in his Affidavit and below.

15. With these conservatisms, the K_{eff} of the pool is normally less than 0.75, and is less than 0.95 (95% probability at 95% confidence level) for the postulated accident condition of no soluble boron in the pool. The soluble boron concentration in the Millstone 2 pool ordinarily is maintained in excess of 1720 ppm.

<u>BORAFLEX DEGRADATION</u>

16. Contention 1, subissue 1, as admitted by the ASLB in its November 24, 1992 Memorandum and Order, states:

> **What is the actual state of the Boroflex [sic] box degradation, and what is the corresponding disposition of the water gaps? <u>Id.</u>, ¶ 8. The Licensee examined approximately half of the poisoned rack cells with a defect rate of 16%.[2] If the sample is not representative, the gaps may be larger than expected, or locally concentrated. A concentration**

of gaps would cause local enhancement of the neutron distribution with an effect of increasing k_{∞}.

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[3]Dr. Kaku incorrectly stated that only 16% of the Boroflex boxes were examined. Affidavit, ¶ 7. The NRC Staff caught this error and noted that the defect rate is 16%. The sampling consisted of approximately half of the poisoned rack cells. . . .

17. As described by Mr. Betancourt in his separate Affidavit, I performed Blackness testing for NUSCo to determine the "actual state of Boraflex degradation," measuring a large fraction (46%) of all the cells containing Boraflex and most of the cells (70%) in which gaps might be expected (i.e., the irradiated cells).

18. Neutron logging, referred to as Blackness testing, uses a sealed neutron source and is intended to measure relative thermal neutron attenuation in the walls of high density spent fuel storage racks of nuclear power plants. The objective of these measurements is to confirm the presence of the neutron absorber material prior to initial operation and/or to periodically verify the continued presence and integrity of the absorber material (e.g., Boral or Boraflex) during long-term storage of spent fuel assemblies. The technique is a derivative of well-logging methods successfully used in the oil industry for many years.

19. During operation, a testing tool containing a Cf-252 source and four BF_3 neutron detectors is lowered into the storage pool and

vertically traverses selected storage cells. The test tool itself is a stainless steel container incorporating a source holder and four thermal neutron detectors, one for each wall. Neutrons from the Cf-252 source pass through the walls of the cell, become thermalized (moderated) in the water of adjacent cells, and diffuse (scatter) back toward the test tool. These back-scattered thermal neutrons are absorbed in areas of the rack walls where the absorber material is present and intact. However, in areas where the absorber material is missing (i.e, gaps) or significantly degraded, the thermal neutrons will pass through and be registered as counts by the detectors inside the testing tool. Increases in thermal neutron counting rates are interpreted as indicating missing or significantly degraded absorber material in the storage cell being measured, while low counting rates confirm that the absorber material is present and intact.

20. Mr. Betancourt described the results of the Millstone 2 Blackness testing campaigns. I conclude that the Blackness testing conducted was a representative sample and provides assurance that the actual state of Boraflex has been conservatively enveloped in the Amendment 158 criticality analysis. A comparison of what has been observed and what was assumed in the analysis proves this point. Dr. Kaku has presented no actual data on Boraflex degradation at Millstone 2 or elsewhere in the nuclear industry to demonstrate that gaps

larger or more clustered than those considered in the criticality calculation might exist at Millstone 2.

21. The criticality analysis assumed 5.65 inch gaps compared to an approximate 0.8 inch average observed gap. The assumed gap also exceeds the largest single observed gap. The largest gap was 2.8 inches, in cell J9, which was removed from the spent fuel pool and replaced with a new Boraflex cell. Cell D9, with its largest gap being 2.2 inches, also was removed and disassembled for study.

22. Boraflex shrinkage, and subsequent formation of a gap, is a function of radiation exposure. With cumulative exposure to radiation in the spent fuel pool, changes to the physical properties of Boraflex cease after the Boraflex has attained a certain radiation dose (called the "Saturation Dose") estimated to be about 5×10^{10} total rads. Some of the Boraflex panels in the Millstone 2 racks have to date attained about 75% of the Saturation Dose. The assumed gaps (5.65 inches) in the criticality analysis for Amendment 158 are larger than would be expected (3.6 to 4.3 inches) for Boraflex panels that have reached the Saturation Dose and in which the maximum size gaps might occur.

23. Next, for purposes of the criticality analysis, we assumed the 5.65 inch gaps occurred in all (100%) panels (there are four

panels in each storage cell), although 87% of all the panels actually tested at Millstone 2 were found to be without gaps. This is a very conservative assumption.

24. The assumption of shrinkage in width was also conservative. The physical examination of Cell D9 removed from the pool did not show any visible evidence of such shrinkage.

25. For purposes of the criticality analysis, the assumed gaps were distributed randomly in the axial direction. Paragraph 4.S-4 on page 2 of Dr. Kaku's March 1993 affidavit identifies a concern relating to the assumption of a random distribution of gaps in the Boraflex. However, contrary to this concern, measurements on numerous racks with gaps in the Boraflex (at Millstone 2 and at other nuclear plants) support the fact that the distribution of gaps is essentially random in the axial direction. Therefore, the assumed distribution is consistent with measured test data. Dr. Kaku provides no contrary data.

26. Prior to submitting Amendment 158, a calculation also was made assuming a hypothetical and unrealistic concentration of gaps over only the central 50 percent (in the axial direction) of the panels. The assumptions regarding the number and size of gaps were the same as in the credited analysis. The resulting K_{eff} for this case was still below the design limit of 0.95.

27. Since reviewing Dr. Kaku's statements, I also have made calculations ignoring the actual observed gap distribution and assuming a hypothetical "clustering" of gaps at the axial center (the location of maximum reactivity effect). The first such analysis assumed a gap (again, 4% or 5.65 inches) in each panel of four contiguous storage cells, with the gaps aligned at the approximate center of the panels. The second analysis was based on the same assumption for every cell in Region B. Results of these hypothetical conditions were as follows:

Gaps clustered in four cells - K_{eff} of 0.9275

Gaps clustered in all cells - K_{eff} of 0.9534

Therefore, only under the extremely unreasonable assumption of "clustering" (i.e., near perfect alignment) of gaps simultaneously at the center in all panels in all cells in Region B, and the concurrent loss of all soluble boron, would the maximum K_{eff} slightly exceed the NRC limit on reactivity. However, even in this case the fuel would be maintained significantly sub-critical. Blackness testing of the cells confirms that neither level of clustering exists in the Millstone 2 racks. Consequently, clustering of gaps is not a genuine issue.

28. Boraflex shrinkage and gap formation also are assumed conservatively in the analysis to result in the removal of poison material. In reality, the total amount of boron is expected to be preserved through an increase in the density of the material. This increase in density is not credited in the criticality analysis.

29. Based on the above, I conclude that this aspect of the contention does not raise a genuine technical issue. The criticality analysis very conservatively bounds actual degradation of Boraflex at Millstone 2. Dr. Kaku has provided no material facts that would challenge this conclusion. His views on the subject to date are mere speculation (i.e., he discusses what he thinks might be the case if we had not adequately bounded the Boraflex degradation).

BENCHMARKING

30. In this proceeding, the contention related to benchmarking the criticality calculations, as set forth by the ASLB, is:

> **To what extent are the benchmark data used by the Licensee representative of the arrangement of Boroflex [sic] boxes, fuel boxes, and water in the storage pool? Id., ¶ 9.**

31. The benchmarking of the KENO model used in the Millstone 2 criticality analysis for Amendment 158 was described in the April 1992 HOLTEC final report provided by Northeast Utilities

to Cooperative Citizens Monitoring Network in the package of discovery materials. Specifically, the model was benchmarked first against critical experiments. Critical experiments used for the benchmarking were selected from the series of B&W critical experiments and were chosen to be as nearly representative of the Millstone 2 racks as possible. Both the critical experiments and the Millstone 2 racks consist of geometrical arrays of boxes into which the fuel assemblies are located, surrounded by thin sheets of absorber material containing boron. (The boron-10 isotope is the strongly absorbing nuclide responsible for neutron absorption.) Other critical experiments, while less representative, were checked and found also to provide good agreement.

32. In addition, the KENO model used in the Millstone 2 analysis was also verified against an independent means of evaluation, in this case CASMO-3. As described in the HOLTEC final report of April 1992, provided to Cooperative Citizens Monitoring Network and Dr. Kaku during discovery, this benchmarking was conducted to verify the NITAWL-KENO-5a methodology with the 27-group SCALE cross-section library for use in criticality safety calculations of high density spent fuel storage racks. Both the KENO and CASMO-3 calculational methods are based upon transport theory, but CASMO-3 solves the neutron transport problem by the use of capture probabilities, a methodology entirely different from that in KENO. The intercomparisons

consisted of a series of CASMO-3 and NITAWL-KENO-5a calculations performed on identical storage cells representative of high-density spent fuel storage racks with thin strongly-absorbing panels. (Intercomparisons between analytical methods are endorsed by NRC Regulatory Guide 5.14.) The results of the intercomparison calculations (shown in Table 3, Appendix C of the April 1992 HOLTEC report) are within the normal statistical variation of KENO calculations.

33. Based on the above, this aspect of Contention 1 is without merit. Furthermore, Dr. Kaku's position, as expressed on page 12 of his March 1993 affidavit, is that "no benchmarking was ever done." This simply is not true.

<u>NUMBER OF NEUTRON HISTORIES AND NEUTRON GROUPS</u>

34. Contention 1, issue 3, as admitted by the ASLB, asks:

> **Have the Monte Carlo calculations incorporated enough iterations to provide a good estimate of the pool's reactivity? <u>Id.</u>, ¶ 10(d).**

Dr. Kaku's affidavits raise questions about the number of neutron groups as well as the number of histories when discussing the adequacy of the criticality calculations. I have assumed that the ASLB reference to "iterations" could mean either, but histories is actually the more likely focus of the original issue.

35. The criticality calculations supporting Amendment 158 rely on the Monte Carlo method of analysis. This method uses the "random walk" technique which simulates the actual behavior of neutrons by tracking neutrons one-by-one through the system until they are absorbed in some material or escape from the system. At each collision, the various reactions are treated stochastically: fission, scattering and radiative capture. The analysis involves calculations with hundreds of thousands of individual neutron "histories," or iterations, until enough histories are accumulated to provide a statistically acceptable description of the system multiplication factor. Random numbers are utilized to determine the process which occurs at each neutron interaction. In his March 1993 affidavit, Dr. Kaku alleges that the neutron equations and the Monte Carlo analyses are unreliable; however, he does not provide any realistic basis for such a conclusion. We will show in the following paragraphs that there is no basis for any of Dr. Kaku's allegations.

36. In the KENO calculations supporting Amendment 158, the minimum number of neutron histories or iterations was 500,000 and, for the reference case, a total of 1,250,000 histories was used. Subsequently, to demonstrate the adequacy of the number of histories, a calculation was made for up to 5,000,000 histories, with results shown below:

50,000 histories: $K_{eff} = 0.917$

125,000 histories: $K_{eff} = 0.922$

250,000 histories: $K_{eff} = 0.921$

500,000 histories: $K_{eff} = 0.923$

1,250,000 histories: $K_{eff} = 0.922$

5,000,000 histories: $K_{eff} = 0.922$

These data demonstrate that the number of histories is more than adequate to provide a good estimate of the pool reactivity. Normally, 125,000 histories would have been adequate to reach convergence. However, to assure convergence in the presence of small volume regions (i.e., gaps) a larger number of histories was used for additional confidence in the calculations.

37. Dr. Kaku, in his affidavits, also discusses the number of "neutron groups" used in the criticality analysis. Normally, the neutron spectrum is divided into energy groups, with each group representing the averaged cross-sections over the energy bounds of that group. In the calculations for Millstone 2, a 27 group cross-section library was used. This library has been demonstrated, by comparison to 218-group models and by benchmarking comparisons, to be more than adequate to represent the Millstone 2 racks. To demonstrate this point, a calculation of the Millstone 2 spent fuel pool criticality was made with a 218-group cross-section library which gave a

maximum K_{eff} of 0.919 compared to a K_{eff} of 0.920 obtained with the 27-group library. This confirms the validity of the 27-group library for criticality safety analyses.

MONTE CARLO/KENO TECHNIQUES GENERALLY

38. Setting aside these specific complaints, Dr. Kaku suggests in his March 1993 affidavit (without citing factual justification) that KENO (and the Monte Carlo technique) itself is inadequate. However, Monte Carlo analysis has been an accepted technique in nuclear analysis for many years and has been referred to as the closest to actual measurement. For example, the following early references illustrate this point:

- H. Kahn, "Stochastic (Monte Carlo) Attenuation Analysis," Rand Report R-163, June 14, 1949.
- U.S. National Bureau of Standards, "Monte Carlo Method," Applied Mathematics Series, No. 12, June 11, 1951.
- G. Goetzel and M.H. Kalos, "Monte Carlo Methods in Transport Problems," in "Progress in Nuclear Energy, Physics and Mathematics," Pergamon Press, New York, 1958.

39. By the time of publication of ANL-5800, the Argonne National Laboratory Reactor Physics Handbook in 1963 and the Naval Reactors Handbook in 1964, the Monte Carlo technique had become an acceptable technique, often used as a reference for other methods of analysis. In the early days, the use of Monte Carlo techniques was severely restricted by the availability of high speed computers. Today, high speed computers are readily

available. The KENO code itself has been used for over 10 years and is being used worldwide as a standard technique for nuclear analysis.

40. Many of Dr. Kaku's observations apparently derive from a confusion between diffusion theory and transport theory. Neutron transport is well described by the Boltzman equation. The only assumptions are that neutron-neutron interactions are negligible and there is no significant decay of the neutrons in the transport process. Both are very good approximations. KENO is well suited for the analysis of complex systems containing thin strong absorbers; it is fully capable of treating the geometry of the Millstone 2 racks, including the thin Boraflex absorber panels. While the neutron flux may decrease significantly in strong absorbers (or at free boundaries) the flux distribution is everywhere continuous and differentiable. Neutrons are not static and a discontinuity in flux or neutron density is not physically possible.

41. Monte Carlo and diffusion theory are different methods of analysis. In diffusion theory, the assumption is made that neutron diffuses in the same manner as a gas and that Fick's equation for gas diffusion can therefore be used to describe the behavior of neutrons. This is a simplified approach that neglects higher order scattering terms, and in general cannot respond properly to steep gradients in neutron flux such as

occurs at the boundary of strong absorbers, unless a transport theory boundary condition is imposed (so-called Blackness theory). Diffusion theory was not used in the Amendment 158 analysis of the Millstone 2 racks. Consequently, references to diffusion theory in Dr. Kaku's statements are not germane to the analysis or to the issues framed by the ASLB.

42. The suitability of transport theory, and KENO in particular, to treat the neutron absorption in the absorber panels is clearly demonstrated by the benchmarking calculations discussed above. This included benchmarking against critical experiments with thin strong neutron absorbers, simulating as realistically as possible the Millstone 2 racks. Furthermore, the independent calculations with CASMO-3 show very good agreement with the KENO calculations in numerous storage racks at many rack installations throughout the world, all with thin strong absorbers. CASMO-3 solves the transport equation by using capture probabilities which is an entirely different methodology than that used in KENO. It would be unreasonable to expect that two completely independent methods of analysis could experience the same calculational error. Both the benchmarking and the comparison with an independent method of analysis confirm the ability of KENO to adequately treat thin strong absorbers.

43. In one portion of Dr. Kaku's second affidavit (Section 11a), page 14, he alleges that the validity of the random number generation methodology used in KENO is inadequate. This allegation evidences a misunderstanding of the nature of both random numbers and computers. Duplicate calculations performed with the same sequence of random numbers on the same problem and on the same computer should give exactly the same result (and is generally required to do so for Quality Assurance purposes). Within any single calculation, the numbers used are a sequence of random numbers. The same sequence may be recalled by choosing the same starting number (called a "seed"). If a different seed is specified, a completely different sequence of random numbers will be generated. Calculations of the Millstone 2 racks with different sequences of random numbers are as follows:

1) 0.9219 ± 0.0007
2) 0.9219 ± 0.0007
3) 0.9218 ± 0.0007
4) 0.9217 ± 0.0007
5) 0.9221 ± 0.0007

These data show that five random number sequences yield the same result within the expected statistical variation.

44. Dr. Kaku's March 1993 affidavit, at page 14, also refers to an unspecified article in the New York Times ("in the past month or so") as indicating errors in random number generators. Upon some research, it is possible that Dr. Kaku was referring to a Times story of January 12, 1993, which in turn refers to a paper in Physical Review Letters of December 7, 1992, by Ferrenburg, Landau and Wong. I have since reviewed that paper. It was a study of random number sequences and reported a very small error (generally in the fourth or fifth decimal places). While the point of the paper may be aesthetically correct, the "error" reported is far too small to have any practical significance to the Amendment 158 criticality analysis. Any inadequacy in the random number generator in the KENO code would have been readily revealed in the many benchmark calculations made on numerous critical experiments and would have appeared in the calculations listed above.

45. In section 10.S-10, on page 13 of his March 1993 affidavit, Dr. Kaku states that "a calculation is only as good as its initial guess." This statement is completely erroneous and betrays a lack of knowledge of the Monte Carlo process as it relates to nuclear analyses of the type performed by KENO. KENO calculations are not in any way dependent upon an initial guess. A starting fission source distribution is input to the program but does not affect the results. To illustrate this, a calculation was made with a grossly distorted starting

distribution (initial fission source assumed to occur in only one corner of the array), and compared with a corresponding calculation with a uniform starting distribution. The result was a K_{eff} of 0.9216 ± 0.0007 for the distorted distribution compared to a K_{eff} of 0.9218 ± 0.007 for the uniform starting distribution. These calculations clearly show that the concern over starting "guesses" is not in any way a substantive issue. Any change in boundary conditions would, of course, define a new and different problem, but such a problem is unrelated to the issue of Amendment 158 analysis.

46. On the last page of his March 1993 affidavit, Dr. Kaku summarizes in six statements his major concerns and defines what he would consider as a reliable calculation. The calculations for the Millstone 2 racks conform to all but one of these points. The remaining item (Item d) is meaningless in the present situation. Again, it would appear that Dr. Kaku has not read the criticality analysis report (provided during discovery) and is unaware of what was actually done in the Amendment 158 analysis.

VERTICAL BUCKLING TERM

47. Contention 1, subissue 4, states that:

> **If a vertical buckling term has been used, has it been used correctly? Id., ¶ 10(c).**

48. A vertical buckling term was not used in the Amendment 158 criticality safety analysis of the Millstone 2 storage racks.

49. Buckling is a term usually associated with diffusion theory and a vertical buckling would relate to the leakage of neutrons in the axial direction. (Even if vertical buckling had been used, it would have amounted to about 0.002 to 0.003 δk, which would almost be negligible for the 136 inches of active fuel in the vertical direction.) Instead, the Amendment 158 safety analyses used a full three-dimensional representation of the storage racks which inherently includes an explicit calculation of the axial leakage. This is described in the criticality analysis report, and it appears that Dr. Kaku is not aware of the calculations or methodology that were employed. Since vertical buckling was not used in the criticality calculations supporting Amendment 158, concerns with vertical buckling are not relevant to the amendment.

CONCLUSION

50. The NRC limit on reactivity is a maximum K_{eff} of 0.95 including uncertainties under the postulated accident condition of the loss of all soluble boron. When related to criticality safety, this limitation provides a substantial margin of safety. Furthermore, the nominal K_{eff} of the storage racks (if filled with fuel of the maximum permissible reactivity) is less than 0.75. Only under an extremely unlikely accident condition of

the loss of all soluble boron, does the K_{eff} approach the maximum calculated K_{eff} value of 0.917. Furthermore, this latter value includes a very conservative allowance for the existence of gaps in the Boraflex substantially larger in size and in greater frequency than has been observed in extensive testing.

51. The information above is true and correct to the best of my knowledge and belief.

Stanley E. Turner

Stanley E. Turner

Sworn and subscribed to before me this 30rd day of April, 1993

Notary Public

My commission expires:
My Commission Expires
September 14, 1995

Professional Experience of

Dr. Stanley E. Turner, PE

EDUCATION:

Georgia Institute of Technology, 1943-1944
University of South Carolina, B.S., Chemistry, 1947
University of Texas, Ph.D., Nuclear Chemistry, 1951

PROFESSIONAL AFFILIATIONS:

Registered Professional Engineer, Florida No. 22862

Member of ANS Standards Committee 8.17 on Nuclear Criticality Safety,

Chairman of the ANS 5.3 and 5.4 Working Groups on fission product release.

Formerly member of the ANS 5 Committee and participated in the formulation of the standard on decay heat.

PROFESSIONAL EXPERIENCE:

HOLTEC INTERNATIONAL, Chief Scientist, July 1987 to present

Responsible for all nuclear analyses, including criticality safety analysis of new and spent fuel storage racks, radiological impact evaluations, shielding calculations, fuel management studies and reactor analyses, and projects of a scientific nature (e.g., special radiological monitoring systems, combustible gas generation and control, radiolytic decomposition and radiation damage effects, fission product inventories and release, and others). Has served as an expert witness in three public hearings (Turkey Point 3, Diablo Canyon, and St. Lucie 2)

Also serves as President of NUSURTEC, INC., (A sister Company to HOLTEC) - Responsible for NUSURTEC operations in neutron logging (Blackness testing), radiography of spent fuel storage racks and for physical measurements on surveillance coupons (i.e., neutron attenuation, Shore A hardness, density, dimensional measurements, neutron radiography and tensile strength tests). Serves as Radiation Safety Officer. NUSURTEC Inc. is licensed to possess a Cf neutron source, radioactive surveillance coupons, and equipment contaminated by use in spent fuel pools.

SOUTHERN SCIENCE OFFICE OF BLACK & VEATCH, Consultant (1977-1987)

Dr. Turner was responsible for a wide range of scientific projects, including criticality analyses of new and spent fuel storage racks, fission product release studies, radiological monitoring systems, assessment of alternate fuel cycles, verification of fuel management analytical methods, radiological assessments, reactor physics studies, and safety analyses/evaluations.

Since 1981, he was involved in the design, evaluation, and licensing of high density spent fuel storage racks, including criticality safety and assessment of radiological consequences. In addition, he has evaluated the core physics performance of testing, research, training, production and power reactors for various government agencies. As Project Manager for several U.S. Arms Control and Disarmament programs, Dr. Turner established export restrictions on graphite purity (for non-proliferation), investigated advanced PWR concepts, including extended fuel burnup (with greater core regionalization), alternate methods of reactivity control, and other possible means of improving fuel utilization.

NUS CORPORATION, Senior Consultant (1973-1977)

Assignments at NUS Corporation included Project Manager responsibility for the assessment of post-LOCA hydrogen generation and methods of control, development of specialized radiological monitoring systems, investigation of radiolytic decomposition of Halon-1301 (bromotrifluromethane) and use of Halon-1301 for fire control (and inhibition of hydrogen burning), fission product inventory and release calculations, tritium production and distribution during reactor operations, reactor physics analysis and production capabilities of a wide variety of reactor types, and special studies among which were a survey of European nuclear fuel cycle plans and capabilities, and a generic review of public issues in the Nation's nuclear power program.

SOUTHERN NUCLEAR ENGINEERING. Inc. Vice President, Physics (1964-1973)

During this period, Dr. Turner managed and/or participated in a number of projects that involved reactor physics calculations, evaluation of heavy element production, review of licensing documents, assessing tritium generation and methods of control, preparation of operating procedures, safety assessments of special purpose reactors, evaluation of consequences of industrial sabotage in nuclear power plants, and studies of maritime reactors. He was also involved in a number of reactor-physics oriented classified projects for the US government.

GENERAL NUCLEAR ENGINEERING CORP., Senior Staff Physicist (1957-1964)

In this capacity, Dr. Turner performed or directed most of the fuel cycle cost evaluations, heavy element production calculations, and fuel management work performed by the company. He planned and coordinated various experiments and testing programs, and managed R&D activities related to advanced fuel element designs. He served as Project Manager for the NUSU (integral superheat) reactor project. In addition, he was a member and Vice-Chairman of the Safety Committee of an operating nuclear power plant and participated in plant licensing actions and safety reviews.

SOCONY-MOBIL RESEARCH LABORATORY, Physicist (1952-1957)

This activity involved setting-up and operating a physics R&D laboratory (including two Van de Graaff accelerators), the design and construction or specification of various radiation measurement systems (including scintillation detector, multi-channel pulse-height analyzers, thermoluminescence devices, G-M tubes and proportional counters) and the performance of various oil well field logging tests and experimental irradiations. Multi-curie radiation sources (Po-Be, Co-60 and Cs-137) as well as shortlived activities from irradiations were used in programs directed toward improved oil-exploration techniques. Dr. Turner holds two early patents on nuclear oil-well logging methods and subsurface activation analysis.

U.S.NAVAL RADIOLOGICAL DEFENCE LABORATORY, Physicist (1951-1952)

In this assignment, Dr. Turner was selected to lead a field unit in a series of atomic bomb tests (Jackass Flats, Nevada) related to fall-out distributions. He designed, participated in the construction, and supervised the operation of a mobile field laboratory, including supporting radiation detection systems, to measure the radiological characteristics and distribution of fission products.

HONORS

Fellow, American Institute of Chemists
Listing in Who's Who of American Men of Science
Sigma Pi Sigma (Physics)
Sigma Xi
Blue Key (Scholastic)
Phi Lambda Upsilon (Chemistry)
Pan American Fellowship

PUBLICATIONS RELATED TO SPENT FUEL STORAGE

Criticality Safety Analyses and Licensing Submittals (or upgrade analyses) for more than 36 plants (Fuel Storage Racks)

S.E. Turner and M.K. Gurley, "Evaluation of AMPX-KENO Benchmark Calculations for High-Density Spent Fuel Storage Racks", Nuclear Science and Engineering, Vol 80, No.2 pp 230-237, February 1982

S.E. Turner, "Nuclear Criticality Safety Considerations in the Design of High Density Spent Fuel Storage Racks, Paper 83-NE-6, presented at the 1983 American Society of Mechanical Engineers Conference, June 1983

S.E. Turner "Storage of Burned PWR and BWR Fuel", American Nuclear Society, Winter Meeting, November 1987

S.E. Turner, "Methods for Neutron Absorber Material Surveillance", presented at the 1988 Joint Power Conference, American Society of Mechanical Engineering, September 1988

S. E. Turner, "Uncertainty Analysis - Burnup Distributions", presented at the DOE/SANDIA Technical Meeting on Fuel Burnup Credit, Special Session, ANS/ENS Conference, Washington, D.C.,November 2, 1988

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

In the Matter of	)	
	)	
NORTHEAST NUCLEAR ENERGY CO.	)	Docket No. 50-336-OLA
	)	(Spent Fuel Pool Design)
(Millstone Nuclear Power Station	)	
Unit No. 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NORTHEAST NUCLEAR ENERGY COMPANY'S MOTION FOR SUMMARY DISPOSITION OF CCMN'S CONTENTION 1," and accompanying affidavits, have been served on the following by deposit in the United States Mail, first class, this 7th day of May, 1993:

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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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