UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

SOUTH CAROLINA ELECTRIC & GAS COMPANY and SOUTH CAROLINA PUBLIC SERVICE AUTHORITY)) Docket No. 395-OL
(Virgil C. Summer Nuclear Station, Unit 1)))

AFFIDAVIT OF LAWRENCE E. HOCHREITER

My name is Lawrence E. Hochreiter, and my qualifications can be found in Attachment A. I am an Advisory Engineer at Westinghouse Nuclear Energy Systems. I have read the affidavit and testimony of Dr. Michio Kaku (Tr. 3573 - 3655, 3670 - 3764). Before discussing Dr. Kaku's affidavit or testimony, I would like to make some general statements:

1.) W uses NRC approved licensing models and computer codes for Loss of Coolant Accident (LOCA) analysis. The codes and models used for the design basis accident conform to the Appendix K criteria. The computer codes used to analyze the design basis transient (LOCA) for V. C. Summer plant met the Appendix K requirements and were approved by the NRC. (Staff SER page 6-23).

2.) Dr. Kaku at several points claims that computer codes have not been verified. The computer codes and models used in the design basis LOCA transients have been verified against test data from separate effects tests such as FLECHT, and integral systems tests of different scale such as semiscale and LOFT. The validations of the codes have been reviewed and approved by the NRC. The Westinghouse

computer models have also been used for blind test predictions in the NRC standard problem program as another method of code verification.

With specific references to Dr. Kaku's affidavit, I offer the following observations. I have used Dr. Kaku's Roman and Arabic numbering system for easy reference.

Sect. I) Evacuation and Accident Hazards At the V. C. Summer Plant

Para. 2) As explained above, the LOCA analyses are performed using evaluation models which satisfy 10CFR50 Appendix K and which have been reviewed and accepted by the NRC. The maximum acceptable temperature using such models was set by 10CFR50.46(b)(7) at 2200°F after extensive rulemaking hearings in which reliability of the computer codes as well as all other known uncertainties were considered. This temperature includes the margin of safety for such uncertainties found to be adequate in the ECCS hearings. The use of these analyses to establish the peaking factor does not constitute retrofitting the data. Dr. Kaku apparently does not understand that with respect to limiting potential peak clad temperature following a loss-of-coolant accident operation of a nuclear reactor is controlled by limiting the peaking factor (peak divided by average linear kilowatts/foot) in the core together with maximum power during normal operation and it is to these limitations which 10CFR50.46(a)(1) refers when it states that conformance with the criteria may require that restrictions be imposed on reactor operation. When one performs a LOCA calculation, the peaking factor is increased until the calculated peak clad temperature equals 2200°F for the maximum power rating. This establishes the maximum acceptable peaking factor under 10CFR50.46. This maximum acceptable peaking factor corresponding to 2200°F and the maximum power level

are include! in the technical specifications which become part of the operating license and with which the operator must either comply or shut down his facility. To the extent that peaking factors do not reach the maximum acceptable valve and calculated peak clad temperatures do not reach 2200oF there is margin over and above the minimum safety margin found to be acceptable in the extensive ECCS hearings.

The design of the emergency core cooling system incorporates suitable redundancy, interconnections, leak detection, isolation and containment capabilities as required by General Design Criteria 35 so that the ECCS safety function is accomplished assuming any single failure. Single failure for this purpose means an occurence which results in the loss of capability of a component to perform its intended safety function with respect to protection against a LOCA and includes multiple failures resulting from a single occurrence. The example cited by Dr. Kaku of a pressurizer PORV failure transforming a major primary cold leg break from a relatively harmless Class 8 accident into a more serious Class 9 accident is without merit. The consequences of a major cold leg break would be unaffected by the failure of a pressurizer PORV. The additional flow area resulting from such a failure would be miniscule in comparison to the area of the double ended break of the 30 inch reactor coolant pipe.

I disagree strongly with the statement indicating the results presented in Chapter 15 of the FSAR are based on pure speculation. Specific accidents are analyzed to see how they challenge the design an the safety system response. Conservative assumptions are made on the response characteristics of the plant such that the design is fully challenged. All these conservative assumptions are designed to make the calculated accident transient more severe, and thus provide increased safety margin.

In summary, I would expect anyone who claims to be familiar with LOCA analysis to be aware of the matters covered in the ECCS Rulemaking and in 10CFR50 Appendix K.

Para. 5e.) As stated earlier the computer codes used for LOCA analysis satisfy 10CFR50.46 and 10CFR50 Appendix K and have been verified and approved by the NRC. With regard to LOFT, LOFT experiments are scaled tests with known differences between the test and a commercial PWR. When the licensing codes are used to model LOFT, the code calculations indicate much higher temperatures than do the experiments indicating that margin exists in the code calculations. Best estimate codes which more accurately model the two-phase flow and do not use the conservative model assumptions specified in Appendix K, compare much more favorably with the LOFT data. Dr. Kaku apparently does not understand how test results are used in LOCA analysis.

Para. 5h.) Dr. Kaku apparently does not understand the calculated large break accident. The emergency core cooling system is designed to recover the core after a large break. During the blowdown phase, emergency core cooling water is initially assumed to be swept out the break with the RCS water. As blowdown continues and the system depressurizes, the ECCS water is calculated to penetrate the downcomer, fill the lower plenum and initiate core reflooding. At this point the conservative Appendix K requirements would show that more ECCS water is injected into the system than can flow up through the core. Therefore, the ECCS water injection flow is larger than the break flow.

Page 8, Item b - Spontaneous Vessel Rupture

Dr. Kaku stated that the British have grave doubts on the purchase of U.S. reactors because of concerns of pressure vessel integrity. After an extensive safety review of the Westinghouse PWR design concept, the British Nuclear Installation Inspectorate found the Westinghouse PWR to be licensable in Great Britain. Any concerns they may have had regarding pressure vessel integrity were satisfactorily addressed. Since that time the British have entered into a licensing and technology exchange agreement with Westinghouse for the introduction of the PWR into Great Britain. I am aware of the British inquiry into this area because I am involved in technology transfer to the licensee under this agreement.

Para. 7.) With regard to TMI-2, Westinghouse performed calculations assuming that the PORV did not close. Our calculations, which used the NSAC-1 flow history and plant data, indicated no fuel melting in the core. Also later, NSAC revised its estimates of core inventory and net inflow to the reactor which resulted in a larger core inventory and greater makeup flows. If these flows would be used in our calculations we would again not predict any fuel melting and the top of the core would benefit from improved steam cooling. We should have expected Dr. Kaku, before making the argument which he does, to have researched the literature and the NRC's public document room in which these calcuations are described.

Para. 8&9.) In my opinion Dr. Kaku does not give a balanced view of the industry response to the accident at TMI-2. The Kemeny Commission did indicate deficiencies in the management of TMI-2 which should be applied to all reactors. The industry has mounted a vigorous program through AIF,

INPO and NSAC to address these concerns. Similar concerns were identified by the NRC in its post TMI reports (NUREG-0660, NUREG-0737) and the industry is working to incorporate the new requirements and regulations. Clearly the industry has responded to learn the lessons of TMI-2.

Para. 11.) With regard to this item; all sides were heard from in the emergency core cooling hearings which lasted two years. Technical concerns expressed by different parties including those quoted by Dr. Kaku were addressed on a technical basis such that a meaningful conclusion could be reached. Also the NRC staff is chartered with the responsibility of assessing the Appendix K requirements to the light of any new experimental data and ensuring that the Appendix K models and assumptions remain valid. In this regard the NRC gave advance notice of a proposed rulemaking on December 6, 1978 but has taken no further action since the close of the comment period. We should have expected Dr. Kaku to have researched whether or not these points were considered (which they were) in the rulemaking.

Para. 12.) The fact that the NRC has a list of unresolved problems is noteworthy since it means that they are fulfilling their role as the industry regulator such that industry generic key issues are identified. The industry can then focus its attention on those issues to resolve the NRC concern. Furthermore, the NRC has taken any necessary action on the items to ensure that no immediate safety concern exists with respect to plants in operation. With respect to Dr. Kaku's comment that "we are a quarter of a century into the nuclear age and still have NRC concerns does not reflect well on the NRC and industry" should be taken with a grain of salt. The turnaround time on a nuclear design in the US is about 12 years. We are building a operational information base on the second generation PWR's in

the US. This operating information is being fed back into the design process to improve existing and future designs. Improvements are back-fitted to existing designs as appropriate. (Pages 1-12 and 1-13 of the Staff SER, NUREG-0717)

With regard to specific comments on Dr. Kaku's testimony on accident analysis. The following should be noted:

tr. 3616 Gadolinium is a poison sometimes added to the UO_2 fuel mixture as an oxide for reactivity control but liquid gadolinium is not added to the coolant as Dr. Kaku appears to believe.

tr. 3616 The melting point of UO_2 is approximately 5100^{o}F where as the melting point of pure uranium metal is only 2069^{o}F. A person reasonably familiar with accident analyses involving potential damage to the core should know which is higher even if he did not know the exact numbers.

tr. 3638 Dr. Kaku has testified that he read Chapter 15 of the V. C. Summer FSAR and the NRC's evaluation of the plant FSAR (tr. 3633) and to be familiar with PWR accident analysis, yet on pgs. 3638 and 3639 he apparently has all of the different accidents confused.

First of all the FLECHT data and reports all refer to the experimental reflood data and resulting heat transfer models which are used for the large break LOCA calculations (design basis accident). The LOFTRAN code Dr. Kaku refers to is not used for the design basis LOCA calculations and has no relationship whatsoever to the FLECHT report and analysis.

LOFTRAN is used for ANS Class I, II and some Class III transients calculations but not for the design basis accident (LOCA) calculations, and as such is not required to meet the Appendix K criteria or approval.

tr. 3675 - 3676

With regard to the Appendix K limit of 2200°F peak clad temperature, the concerns expressed by Dr. Kaku on the zirconium-water reaction effects were identified and addressed in the two-year emergency core cooling hearings. The Appendix K rule requires the use of the Baker - Just correlation. Data presented at the core cooling hearings indicated that 2200°F was a conservative limit, acceptable in terms of the zirconium-water reaction effects. Newer experimental data has shown this calculated reaction rate to be conservatively high by approximately a factor of one-third. I would have expected Dr. Kaku to have been familiar with the literature on this subject.

tr. 3678: With regard to a full scale ECCS test: A full scale test can be less demanding on the system being tested than a well devised series of component tests coupled with a good program of analytical predictions of the effect of system failures on components. In a full scale test, one takes what one gets. One has little or no control of influences a particular component will experience. In individual tests, one can subject individual components to conditions two, three, or even ten times worse than they might receive in the full scale test. It is also true, of course, that a single full scale test provides information for only one set of conditions. On the other hand, a series of well devised engineering tests of individual components and groups of compopnents can cover a broad range of accident conditions. Furthermore, the results of these tests can be matched up with a thorough analytical program allowing confident prediction of the results of system failures under a wide variety of conditions. Here again, Dr. Kaku does not demonstrate an understanding of the role of test data.

tr. 3682 - 3683

The NRC safety concerns list has been discussed earlier in the affidavit. (See Page 6 of this document.)

tr. 3694 - 3695, 3697, 3698

Its apparent that Dr. Kaku does not understand the heat transfer modes of a PWR in either steady state operation or during calculated transients in spite of the fact that he has spent 1/3 of his time on accident analysis. In Chapter 15 of the FSAR, for the Class I, II and III accidents the main concern is on DNB (departure from nucleate boiling) since this can lead to a loss of fuel rod cooling capability and could also lead to fuel failure. The initials DNB and the concept of departure from nucleate boiling are so fundamental as to be familiar to anyone involved in accident analysis. Dr. Kaku should have also known that nucleate boiling is preferred cooling mode as compared to film boiling which occurs after DNB.

With regard to steam binding, we do not regard it as a uncharted area of thermal hydraulics since it only concerns calculations of heat transfer in the steam generator and pressure drops in the reactor coolant loop. These matters were thoroughly examined in the emergency core cooling hearings and conservative Appendix K type calculations can be adequately performed in these areas to maximize the steam binding effects and thereby increase the calculated peak clad temperature. Dr. Kaku should have been familiar with the outline of these calculations since they are discussed in the FSAR, NRC's SER and the Appendix K rule. Dr. Kaku's publication in the Technology Review is an editorial covering a broad range of issues. It does not indicate the depth of understanding of either thermal hydraulics or accident analysis one would expect of one to be relied upon in these areas. Dr. Kaku implies that there is some sort of closed network of scientists in

national laboratories who prevent him from getting his articles published in scientific journals. In my experience in conducting reviews of papers being published in my areas, the objective is to ensure that the paper is factual and the resulting data supports the paper's conclusions.

tr. 3698 Dr. Kaku should have been aware of the DNB margin for Class I, II, and III transients, if he read Chapter 15 of the FSAR.

tr. 3698: Dr. Kaku's lack of knowledge on what peak linear heat rate (kilowatt/foot) and hot channel factors are further demonstrates his lack of essential knowledge in the thermal-hydraulics and accident analysis area. The key parameter in the large break LOCA calculations is the peak liner heat rate for the hot channel. It is this channel which reaches 2200°F for the calculated LOCA.

tr. 3730 - 3731

Again Dr. Kaku refers to WCAP - 7907 which is the LOFTRAN report as being necessary for the Appendix K calculations. The LOFTRAN code is <u>not</u> used in the LOCA Appendix K calculations for Westinhouse PWR plants. Nowhere in this report is it stated that it is used for Appendix K analysis. As clearly pointed out by the NRC on page 15-10 of the Staff SER NUREG-0717, the reports WCAP-7907 and WCAP-9230 describe the models used in the evaluation of feedwater system pipe breaks and not loss-of-coolant accidents in the reactor coolant system.

Dr. Kaku refers to the FLECHT program on pg. 3730 as a concern. It should be noted that the FLECHT program and data was fully debated in the Emergency Core Cooling hearings and was approved for use in the Appendix K analysis.

Dr. Kaku also ties the Flecht program to WCAPS - 7907 (LOFTRAN and WCAP - 9230 feedline rupture). There is no connection. Both quoted WCAPS are for Class I, II and III transients which are not design basis LOCA's, whereas the FLECHT tests were designed to understand reflood for a large LOCA.

tr. 3742: Dr. Kaku infers that the thermal-hydraulics of a Class 9 transient can be obtained from a standard text book. Nothing could be further from the truth. The Class 9 Accident is an extremely complex coupled heat transfer, fluid mechanics, and materials problem which requires a high level of expertise in a wide range of disciplines.

Anyone can postulate accident scenarios however to establish the extent to which various accident scenarios are credible is an entirely different matter. In order to determine the credibility of any given event one must have access to detailed knowledge in all the related technical disciplines. Therefore, when one evaluates the credibility of nuclear accident scenarios one must have the assistance of others who have detailed familiarity with just about all the technical disciplines involved in the design, construction and operation of a nuclear plant. For example, taking the simple assumption of a pipe break, to establish its credibility one would have to know the details of the provisions for leak detection, and the sensitivity of those provisions (Fluid Systems and Instrumentation Engineering). Then one would have to know how large a crack would have to be in order to have sufficient leakage to be detected (Thermo-Hydraulics, Radiation Effects, and Fluid Systems Engineering).

Then one would have to know the crack size which could cause rupture of the pipe (Stress Analysis, Metalurgical Engineering and Fracture Mechanics). Then one would have to know the length of time for the smallest detectable crack to grow to the critical size which could cause

rupture under the existing stress fields during operation (Stress Analysis, Systems Engineering, Metalurgical Engineering). Only then would a comparison between leak detection time and the time for a crack to propagate to rupture allow a determination of the likelyhood of the operator detecting the leak and shutting down (Detailed knowledge of Technical Specifications and Operating Procedures) before the crack could propagate to the critical size which could cause rupture. Even after the credibility of the size rupture is established it is necessary to determine the potential effects of such a rupture and the credibility of those effects. This requires a knowledge of the function of the pipe in connection with other systems (Fluid Systems Engineering) and the provisions within the plant to mitigate or prevent adverse effects (control and protection and Nuclear Safety Engineering). Then one would have to evaluate the consequences of each scenario (Fluid Mechanics, Thermo-Hydraulics and Mechanical Effects Analysis) and the credibility of the consequences of any overall scenario would be the combined credibility of all the events required to occur in order to lead to those consequences. While one can postulate scenarios and run computer programs to generate consequences based on the assumed scenario, this of itself will not provide any insight to the credibility of the consequences calculated.

ATTACHMENT A

Educational and Professional Qualification

of

Lawrence E. Hochreiter

Advisory Engineer-Safeguards Engineering

Nuclear Safety Department

Westinghouse Nuclear Energy Systems

Education

B.S. Mechanical Engineering, University of Buffalo, 1963
M.S. Nuclear Engineering, Purdue University, 1967
Ph'D Nuclear Engineering, Purdue University, 1971

Westinghouse short courses on management techniques
MIT short course on two-phase flow and heat transfer, 1973

Professional Experience

From 1963 to 1971 I worked on my M.S. and Ph'D in Nuclear Engineering in the areas of liquid metal heat transfer and turbulent flow fluid mechanics. I also taught a undergraduate course on heat, momentum, and mass transfer.

From 1971 to the present I have been associated with Westinghouse Nuclear Energy Systems and have held different technical and managerial positions of increasing responsibility. From 1971 to 1972 I was a senior engineer in Thermal-Hydraulic Design and worked on developing thermal hydraulic design methods for PWR cores. From 1972 to 1977 I was manager of the Safeguards Development group who had the responsibility of planning, analyzing, and utilizing thermal-hydraulic data from Westinghouse and other experiments to aid in the development and verification of thermal-hydraulic models for Westinghouse safety analysis codes. Programs which were under my direction were the FLECHT program, FLECHT-SET program, steam Water Mixing program and other Internal Westinghouse programs. I was appointed Advisory Engineer in 1977 and am the principal investigator on the NRC/EPRI/Westinghouse FLECHT-SEASET program. I also participated in support efforts for the TMI-2 Accident, Westinghouse Kemeny Commission support, and the Industry Task Force on TMI-2 clean-up activities.

Since 1976 I have been an adjunct professor in Nuclear Engineering at Carnegie-Mellon University and have taught or team taught graduate level courses in thermal-fluids design of Nuclear Reactors, Advanced Topics in Nuclear Safety and Two-Phase flow and Heat Transfer. I am a member of the ASME, K-13 (Neutronics Heat Transfer Group), and the Pittsburgh section of the ANS. I am a reveiwer for the Nuclear Safety Journal, ASME, and Journal of Nuclear Technology. A list of my publications is attached.

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I heareby certify that the foregoing information is true and correct to the best of my knowledge and belief.

Lawrence E. Hochreiter

Lawrence E. Hochreiter

Subscribed and sworn to before me this 6th day of August 1981.

[illegible]

Notary Public

My commission expires: [illegible] Notary Public
[illegible] Borough, [illegible] County
[illegible] Apr 15, 1982
[illegible]