UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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| In the Matter of | ) | Docket Nos. 50-250 OLA-1 |
| | ) | 50-251 OLA-1 |
| FLORIDA POWER AND LIGHT COMPANY | ) | |
| | ) | ASLBP No. 84-496-03 LA |
| (Turkey Point Nuclear Generating Units 3 & 4) | ) | |

AFFIDAVIT OF EDWARD A. DZENIS

I, Edward A. Dzenis, being duly sworn, say as follows:

1. I am Manager of Thermal and Hydraulics Design for the Nuclear Fuel Division of Westinghouse Electric Corporation. My business address is Westinghouse Electric Corporation, Monroeville Mall Office Building, P. O. Box 3912, Pittsburgh, PA 15230. A summary of my professional qualifications and experience is attached hereto as Exhibit A, which is incorporated herein by reference. I have personal knowledge of the matters stated herein, and believe them to be true and correct. This Affidavit is offered in support of "Licensee's Motion for Summary Disposition of Intervenors' Contention (d)."

2. Intervenors' Contention (d) states:

> The proposed decrease in the departure in the nucleate boiling ratio (DNBR) would significantly and adversely affect the margin of safety for the operation of the reactors. The restriction of the DNBR safety limit is intended to prevent overheating of the fuel and possible cladding perforation, which would result in the release of fission products from the fuel. If the minimum allowable DBNR [sic] is reduced from 1.3 to 1.7 [sic; read 1.17] as proposed, this would authorize operation of the fuel much closer to the upper boundary of the nucleate boiling regime. Thus, the safety margin will be significantly reduced. Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the departure from the nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. Thus, the proposed amendment will both

> significantly reduce the safety margin and significantly increase the probability of serious consequences from an accident.

In considering this contention, it is helpful to begin with a basic discussion of core configuration and reactor operation.

3. The Turkey Point Unit 3 and Unit 4 cores are comprised of an array of fuel assemblies, each assembly consisting of a 15x15 array of Zircaloy clad fuel rods. The fuel rods are maintained at safe operating temperatures by the axial flow of reactor coolant water in the channels between adjacent rods.

4. Turkey Point Units 3 and 4 previously operated with Westinghouse 15x15 low-parasitic (LOPAR) fueled cores. Starting the the Turkey Point Unit 3 Cycle 9 and Unit 4 Cycle 10 reloads, both units were refueled with 15x15 optimized fuel assembly (OFA) regions supplied by the Westinghouse Electric Corporation. Future core loadings will range from approximately a 1/3 OFA-2/3 LOPAR mixed core to eventually an all OFA fueled core.

5. Fission products are produced in the fuel during operation. These fission products are normally retained within the fuel matrix and inside the Zircaloy cladding that surrounds each fuel rod. Any fission products which may be released from the fuel rods enter the reactor coolant where they are prevented from escaping to the environment by the reactor coolant pressure boundary which completely encloses the reactor coolant system. Both the level of radioactivity in the reactor coolant and the rate of leakage from that system are monitored and controlled.

6. It has always been considered prudent to take steps to ensure the integrity of all the multiple barriers to fission product release to the environment. In this regard, 10 CFR Part 50, Appendix A requires, in General Design Criterion (GDC) 10, "Reactor Design," that "the reactor core and

associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." With respect to fuel performance, the Nuclear Regulatory Commission (NRC) Staff has prescribed, in the Standard Review Plan, NUREG-0800, Section 4.4, "Thermal and Hydraulic Design," pp. 4.4-2 to -3, July 1981, that these requirements can be met through the use of heat transfer correlations based on experimental data in safety analyses and in establishing technical specifications which assure with 95% confidence that there is a 95% probability that fuel design limits, including departure from nucleate boiling, will not be exceeded.

7. The relationship between fuel performance, in terms of cladding integrity, and heat transfer can be understood by examining temperature as a function of heat flux from the surfaces of the fuel rods for a given system pressure, coolant velocity and temperature. Heat flux is the amount of heat transferred from a surface per unit time per unit area, typically expressed in Btu/hr-ft$^2$. This relationship is shown in Figure 1, which is attached hereto and incorporated herein by reference. As indicated in the figure, the heat flux increases slowly as the rod temperature is increased at low values. In this temperature range, between points A and B in Figure 1, heat is transferred to the coolant water by a process called Forced Convection with no boiling occurring.

8. As the surface temperature of the fuel is increased further, a point is eventually reached where bubbles of steam begin to form on the surface of the fuel rods. This corresponds to point B in Figure 1, and is a form of boiling called Nucleate Boiling. As the bubbles are formed, they are carried away from the rods and into the bulk coolant as a result of the turbulence of

the water. However, the bubbles will soon condense to the liquid state and disappear from the coolant. There is no net production of steam under these circumstances, and the boiling process is termed Subcooled Nucleate Boiling or Local Boiling. If the average temperature of the coolant water reaches its saturation temperature (the temperature at a specific pressure at which water and steam can co-exist in equilibrium), the bubbles continue to be swept away but persist within the flow of water. A net output of steam commences, and the system is now said to be undergoing Saturated Nucleate Boiling or Bulk Boiling.

9. With the onset of Nucleate Boiling, heat moves rapidly into the water, and with increasing surface temperature in the region between B and C in Figure 1, heat transfer is more efficient than between A and B. The heat transfer mode of Nucleate Boiling is very effective for cooling fuel rods. Operation within the Nucleate Boiling mode assures that cladding damage will not occur.

10. If the temperature of the surface of the fuel is increased in the Nucleate Boiling region, adjacent bubbles will eventually coalesce and begin to form a steam film over the surface of the rods. At this point (point C in Figure 1), the system is said to be in a condition leading to a Departure from Nucleate Boiling (abbreviated DNB). The heat flux at the beginning of DNB is called the Critical Heat Flux, (abbreviated CHF).

11. With the beginning of DNB the heat flux into the coolant water begins to drop. This occurs as a result of the heat being forced to pass through the steam into the liquid coolant by less efficient heat transfer mechanisms (i.e. the steam acts as insulation) over the regions of the rods covered by a steam film. The heat flux continues to drop (as indicated by the dotted lines

in Figure 1) with increasing fuel temperature as the total area of the film covering the fuel increases. In this region of Figure 1, the heat transfer process is called Partial Film Boiling.

12. Eventually, when the rod surface temperature is high enough, DNB is complete and the vapor film covers the entire rod. Heat flux to the coolant falls to a minimum value (point D). Beyond this point, any increase in temperature leads to an increase in the heat flux simply because heat transfer through the film, though a poor and inefficient process, nevertheless increases with the temperature difference across the film. The heat transfer process, in this case, is called Full Film Boiling.

13. The existence of the various boiling regimes and DNB is an important consideration in the design of a water cooled reactor. If the reactor power is increased in a region so that the heat flux into the water rises above the CHF, partial film boiling will immediately begin in this channel. However, as explained above, the formation of the film will impede the transfer of heat to the coolant. As a consequence, the heat confined, so to speak, within the fuel will raise the fuel temperature and the surface temperature of the rods forming the channel. This, in turn, leads to an increase in the area of the film, which leads to a further decrease in the heat flux, a further increase in rod surface temperature, and so on. In this way, the wall temperature rapidly increases along the boiling curve between points C and E. These are occurrences which should be prevented. For this reason, it is important to determine the value of CHF, and to keep a reactor from operating near the DNB point.

14. The reactor is kept from operating near the DNB point by ensuring that the heat flux in the reactor is always below the CHF. For this purpose, it is convenient to define a number, called the DNB Ratio (DNBR), as:

$$\text{DNBR} = \frac{\text{Critical Heat Flux}}{\text{Actual Heat Flux}} = \frac{\text{CHF}}{\text{AHF}}$$

In this ratio, CHF is the critical heat flux computed as a function of position along the hottest coolant channel from the appropriate DNB correlation, and AFH is the actual surface heat flux at the same position along the channel. By defining a limit on the minimum DNBR, corresponding to a 95% probability that CHF will not be reached with a 95% confidence level for any particular DNB correlation, the requisite assurance is established that adverse heat transfer conditions will not be reached anywhere in the reactor.

15. Numerous correlations have been developed in the nuclear power industry to predict the occurrence of CHF for various operating conditions and core geometries. These correlations are developed by using the results of tests performed at reactor operating conditions to determine relationships between CHF and various engineering parameters such as coolant temperature, pressure and flow velocity.

16. Two Westinghouse correlations approved by the NRC for determining CHF have been used for Turkey Point Units 3 and 4. A modified W-3 based DNB correlation is approved for use in the analysis of LOPAR fuel. The WRB-1 DNB correlation is approved for use in the analysis of OFA type fuel.

17. The original W-3 correlation was developed conservatively from early experimental studies of DNB conducted with fluid flowing inside heated tubes. The W-3 correlation was subsequently modified to apply to test results of the LOPAR reactor fuel design used at Turkey Point. These modifications were accepted by the NRC to be suitable for reactor rod bundle design and safety analysis. The modified W-3 correlation approved for LOPAR fuel is referred to as the L-grid correlation.

18. Following the development of the W-3 and L-grid DNB correlations, Westinghouse developed a new correlation based exclusively on the large amount

(over 1,100 data points) of CHF data obtained from rod bundle tests. These tests were carried out using arrays of heated rods with cooling water flowing between them, closely simulating the actual fuel rod geometries and conditions of operating PWR's. This large body of data points encompasses all possible combinations of operating reactor conditions occurring during any condition of normal operation, including the effects of anticipated operational occurrences. Each data point represents a measurement of the CHF at a specific set of operating conditions. By basing this correlation, called the WRB-1 correlation, exclusively on rod bundle data, rather than on the single tube data originally used with the W-3 correlation, a more accurate representation of actual operating reactor conditions is obtained.

19. As was previously stated, reactors must be designed in such a way that there is adequate transfer of heat from the fuel rods to the cooling water so that fuel damage is not expected to occur during normal operation, including the effects of anticipated operational occurrences. The NRC has specified that this design basis is met by providing assurance that with 95% confidence there will be at least a 95% probability that the hottest fuel rod in the core does not experience DNB. Specific events which must meet this DNB design basis, and which form a part of the licensing basis for Turkey Point are presented in the Turkey Point Final Safety Analysis Report (FSAR), Section 14. They are:

- Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from a Subcritical Condition
- Uncontrolled RCCA Withdrawal at Power
- RCCA Drop
- Chemical and Volume Control System Malfunction
- Startup of an Inactive Reactor Coolant Loop

- Reduction in Feedwater Enthalpy Incident
- Excessive Load Increase Incident
- Loss of Reactor Coolant Flow
- Loss of External Electrical Load
- Loss of Normal Feedwater
- Loss of Offsite A.C. Power
- Rupture of a Steam Pipe (Valve Malfunction)

20. The DNB design basis is met by specifying a minimum DNBR acceptance limit. The reactor is then designed in such a way that the minimum value of DNBR at any point in the core during normal operation, including anticipated operational occurrences, will be greater than this acceptance limit.

21. The DNBR acceptance limit for a specific DNB correlation is determined as follows:

A. As previously described, the DNB correlation is based on a large body of data obtained from tests in which the heat flux at the point of occurrence of DNB is measured (for example, this consisted of over 1,100 data points for the development of the WRB-1 correlation).

B. NRC-accepted analyses, which model the heat transfer and flow distributions characteristic of a nuclear reactor, utilize the correlation to predict values of CHF given the conditions of each test.

C. These values of predicted CHF are compared with measured values of CHF from the tests. If the correlation developed in Step A were a perfect predictor of DNB for each test condition, the design DNBR would be equal to 1.0. Because differences due to uncertainties are observed between measured and predicted CHF values, NRC-approved statistical methods are used to establish a minimum DNBR acceptance limit to ensure that the design basis (a 95% probability with a 95% confidence level that the hottest rod does not experience DNB) is met.

22. The minimum DNBR acceptance limit required with the use of the L-grid correlation has been statistically determined to be 1.30. This acceptance limit accounts for uncertainties involved in the prediction of DNB with the L-grid correlation.

23. As noted above, the WRB-1 correlation is based strictly on data from rod bundle tests, whereas the L-grid correlation is based on single tube data. The fact that the WRB-1 correlation is a better predictor of DNB for actual nuclear reactor geometries is shown by the result that the minimum DNBR acceptance limit required with the use of the WRB-1 correlation is only 1.17. The WRB-1 acceptance limit was calculated using the same statistical methods as were used in calculating the L-grid DNBR acceptance limit. The 1.17 DNBR acceptance limit has been accepted by the NRC as meeting the DNB design basis when using the WRB-1 correlation.

24. The change in minimum DNBR for the different correlations in no way implies a reduction in the safety margin of a nuclear reactor. This is because the DNB design basis, <u>i.e.</u>, 95% probability with a 95% confidence level that the hottest rod does not experience DNB, remains unchanged. Rather, it demonstrates a natural progression in the understanding of this complex phenomenon as more data is obtained.

25. The purpose of DNB analysis is to confirm that the minimum DNBR value will be above the DNBR acceptance limit as required. DNBR analysis for Turkey Point was performed according to the following procedure:

A. The analytical methods and computer programs used to determine fluid conditions for both fuel types during normal and anticipated transient operation are described in Chapters 3 and 14 of the Turkey Point Unit 3 and 4 FSAR. These same analytical methods and computer programs have been employed for this purpose in safety analyses accepted by the NRC for Turkey Point throughout its operating lifetime including the initial operating license review, and have also been accepted for other Westinghouse plant applications. The most recent NRC Staff approval of safety analyses employing these methods and computer programs is provided by the Safety Evaluation Report (SER) for Amendments 99 and 93 for Turkey Point Units 3 and 4, respectively, dated December 23, 1983.

B. These conditions were then used as input to the applicable DNB correlation (L-grid for LOPAR fuel, WRB-1 for optimized fuel) to determine DNBR values throughout the reactor core.

C. The minimum DNBR value was determined and verified to be greater than the applicable DNBR acceptance limit (1.30 for LOPAR fuel, 1.17 for OFA fuel).

26. Analyses performed for Turkey Point in support of amendments first noticed in the Federal Register on October 7, 1984 -- providing, among other things, for increasing the hot channel factor $F_{\Delta H}$ limit -- demonstrated that the minimum calculated DNBR values for both fuel types are above the DNBR acceptance limit. This was verified for the events which must meet the DNB design basis.

27. It should be noted that although $F_{\Delta H}$ does have a direct impact on calculated DNBR values, the changes of the $F_{\Delta H}$ amendment do not reduce DNBR values to a point where they are below the acceptance limit. Previous DNB analyses (prior to the $F_{\Delta H}$ amendment) showed that the minimum DNBR values of both transient and normal operation not only met the DNB acceptance limit, but in actuality were greater than the acceptance limit by an amount which will be called the "DNBR Available for Design Flexibility." The NRC design basis that there is a 95% probability with 95% confidence that the hottest rod does not undergo DNB defines the safety margin. Figure 2, which

is attached hereto and incorporated herein by reference, shows the relation between this safety margin, "DNBR Available for Design Flexibility," and the effect of increasing $F_{\Delta H}$. It can be seen that the full safety margin is maintained, although there is a change in the "DNBR Available for Design Flexibility" due to an $F_{\Delta H}$ increase.

28. The following conclusions can be made concerning the Turkey Point DNB analysis performed in support of the amendments first noticed on October 7, 1984, including a change in $F_{\Delta H}$:

A. Appropriate NRC-approved methodology has been used in all analyses. Computer programs and DNB correlations used in the analysis were appropriate and NRC approved.

B. There has been no reduction in safety margin. The DNB design basis requires a 95% probability with 95% confidence that the hottest rod does not undergo DNB. This design basis has been met for Turkey Point LOPAR and OFA fuel by meeting their respective DNBR limits (of 1.3 and 1.17).

C. Results of the DNB analysis show that all applicable regulatory requirements have been satisfied.

Further deponent sayeth not.

Edward A. Dzenis

Edward A. Dzenis

STATE OF Pennsylvania

ss.

STATE OF Allegheny

Subscribed and sworn to before me this 8th

day of August, 1984.

My commission expires: 12-14-87

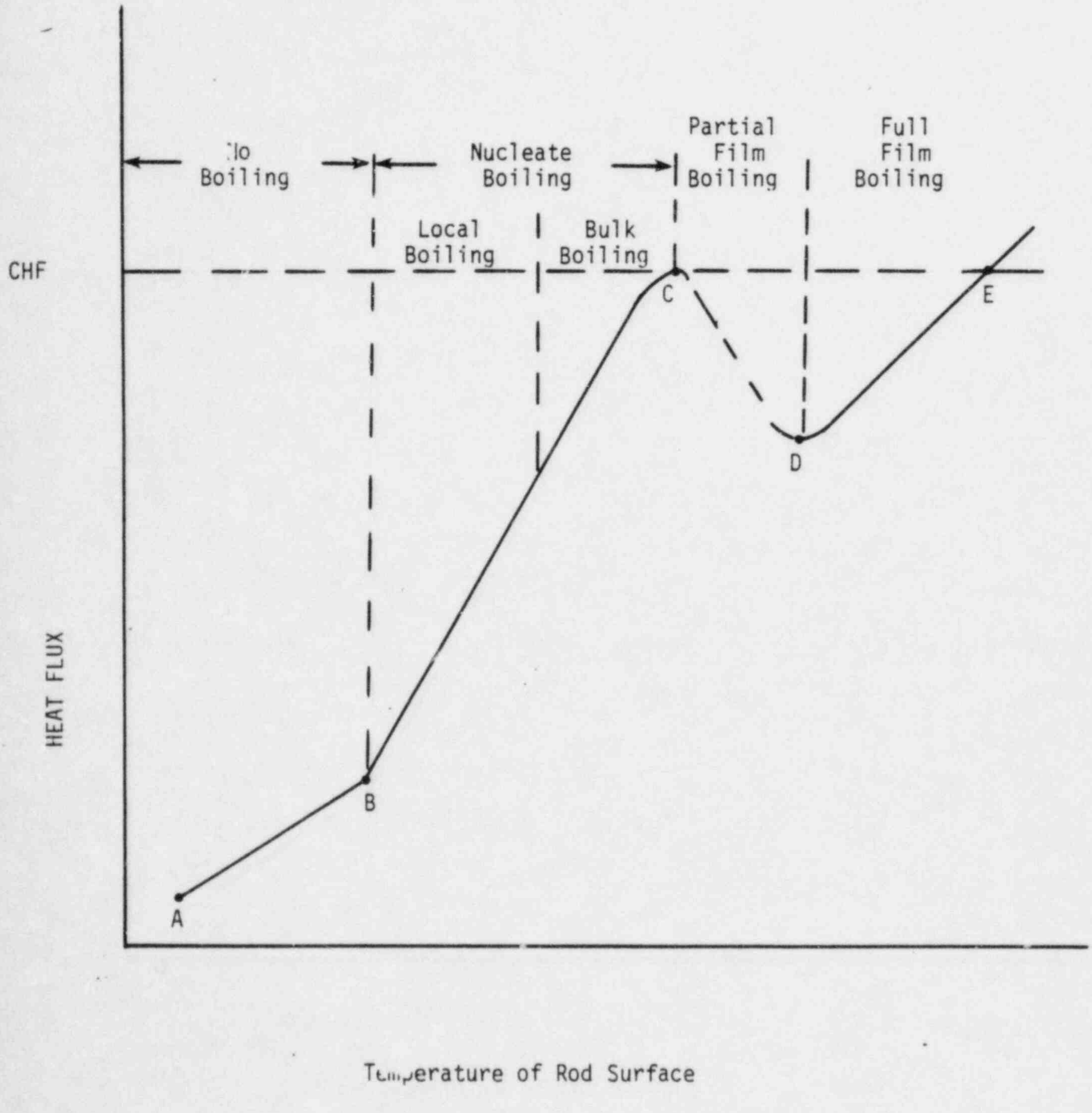
Lorraine M. Piplica

NOTARY PUBLIC

LORRAINE M. PIPLICA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES DEC 14, 1987
Member, Pennsylvania Association of Notaries

<u>Figure 1</u>

Results of an Experiment Showing the Relation of Heat Flux with Rod Surface Temperature



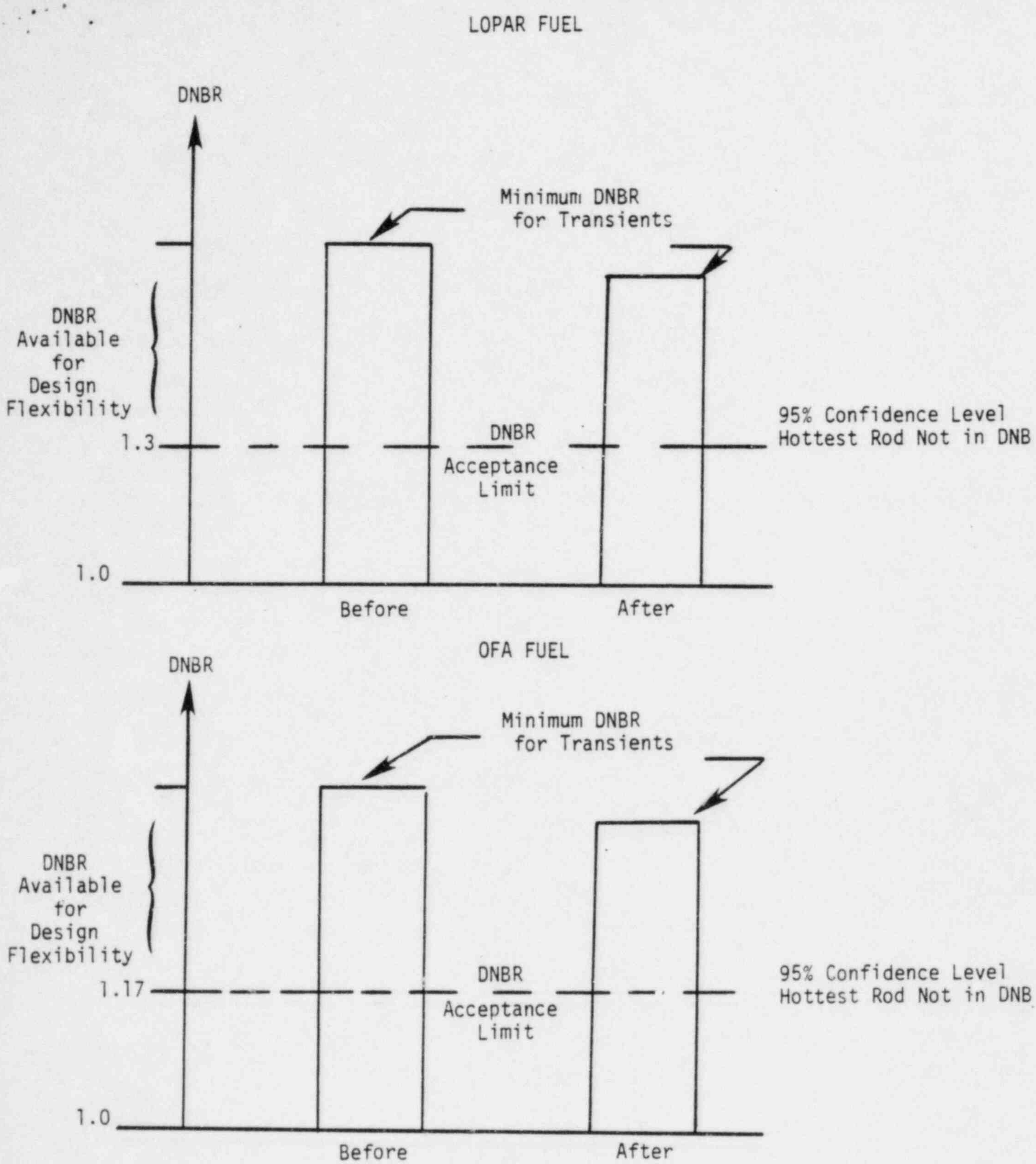



Figure 2

Effect of $F_{\Delta H}$ Increase on DNBR Available for Design Flexibility

Exhibit A

Professional Qualifications and Experience of

Edward A. Dzenis

My name is Edward A. Dzenis and my business address is P. O. Box 3912, Pittsburgh, Pa. 15230. I am employed by Westinghouse Electric Corporation ("Westinghouse") as Manager of Thermal-Hydraulic Design in the Nuclear Fuel Division.

I graduated from Lehigh University with a Bachelor of Science Degree in Mechanical Engineering in May, 1974. While employed by Westinghouse I graduated from Carnegie Mellon University with a Master of Science Degree in Mechanical Engineering in May, 1977. I am currently a Registered Professional Engineer in the Commonwealth of Pennsylvania (certificate number PE-027744-E).

In June, 1974, I joined Westinghouse in the Nuclear Fuel Division of the Water Reactor Division as an Associate Engineer. My duties in the Thermal-Hydraulic Design Department included the analysis of heat transfer and fluid flow aspects of reactor fuel assemblies and related components for pressurized water reactors. These analyses included the determination of core operation limits to insure margin for prevention of departure from nuclear boiling (DNB) and other safety criteria. The results of various postulated accidents were analyzed to check whether these core limits met requirements. I was also responsible for preparing related documentation for submittal to regulatory authorities.

Since that time I have had assignments of increasing responsibility in Thermal-Hydraulic Design and was promoted to the position of

Engineer in August 1976, and Senior Engineer B in April, 1980. In October, 1981, I was promoted to my current position of Manager, Thermal-Hydraulic Design, with responsibility for the efforts of several engineers and technicians in the thermal-hydraulic analysis of fuel for Westinghouse supplied pressurized water reactors including the Turkey Point units. The analysis for the submittals for the increased F-delta-H at Turkey Point Units 3 and 4 was performed under my supervision.