CANADA

PROVINCE OF QUEBEC

DISTRICT OF MONTREAL.

AFFIDAVIT OF GORDON D. J. EDWARDS

I, Gordon Edwards, being duly sworn, say as follows:

1. I am President of the Canadian Coalition for Nuclear Responsibility (Inc.) and Professor of Mathematics and Science at Vanier College (Montreal). My business addresses are, respectively, P.O. Box 236, Snowdon Post Office, Montreal, Quebec, Canada, H3X 3T4; and 5160 Decarie Boulevard, Montreal, Quebec, Canada, H3X 2H9. 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 Ms. Joette Lorion's Intervention, specifically contentions (b) and (d), in the matter of Florida Power and Light (the Licensee), concerning the Turkey Point Nuclear Generating Station, Units 3 & 4.

2. To support the Licensee's integrated program for vessel flux reduction, in order to resolve the pressurized thermal shock issue, the Licensee has decided to change the core configuration -- moving from the present regime involving LOPAR ("low parasitic") fuel to a new regime involving OFA ("optimized fuel assembly") fuel. During the transition period, a "mixed core" of LOPAR fuel and OFA fuel will be used. At the same time, Licensee has applied for permission to offset the reduction in volume of fuel by increasing the hot channel $F_{\Delta H}$ limit from 1.55 to 1.62, and by increasing the total peaking factor F_Q limit from 2.30 to 2.32, thereby obtaining increased power from the lower fuel density. This change in design will increase the fuel temperature in certain parts of the reactor core.

3. Running the fuel at a hotter temperature materially increases both the probability and the consequences of cladding failure, whether there is a loss-of-coolant accident (LOCA) or not. This increase in probability results from the fact that the margin of safety is materially reduced by operating the fuel at a higher temperature. At a certain critical temperature, if there is no LOCA, an insulating film of steam will begin to form around the hot cladding surrounding the fuel ("departure from nucleate boiling" or "DNB"), thereby sharply reducing the rate of heat transfer to the coolant, leading to overheating and rupture of the cladding, which in turn releases fission products (mainly radio-iodines and radioactive noble gases). If there is a LOCA, the same kind of damage can be done by failing to rewet the fuel fast enough. In either case, the time required to reach the critical temperature (following an abnormal occurrence) will be shorter if the initial temperature of the fuel is higher. Moreover, the amount of radio-iodine available to be released in the event of a

cladding failure is directly related to the operating temperature of the fuel prior to the cladding failure; by running the fuel at a higher temperature, more radio-iodine is in the "gap" between the cladding and the fuel, and therefore the consequence of a cladding failure is worse (in turns of the quantity of fission products available to be released).

4. The Nuclear Regulatory Commission has acknowledged that there will be a "reduction in safety margin resulting from the increase in the $F_{\Delta H}$ and F_{Q} limits" [Federal Register, Vol.48, No. 196, page 45862], but argues that there are "no *significant* hazards" involved in granting the Licensee's request. The basis for this judgment is a safety evaluation carried out by the Licensee which calculates peak clad temperatures following hypothetical LOCAs to be "within the maximum limit of 2200°F", and which also calculates that the "departure from nucleate boiling ratio" or "DBNR" has an additional margin of safety that had previously not been identified.

5. Licensee's Motion for Summary Disposition of Intervenor's Contention (b) states, in part:

> Section 50.46 of Nuclear Regulatory Commission regulations requires that an ECCS analysis be performed with an acceptable evaluation model, and result in a calculated maximum fuel element cladding temperature not greater than 2200°F.... ECCS evaluation model analysis utilizing the BART code results in a calculated fuel rod peak clad temperature ("PCT") of 1972°F for a homogenous core of either low-parasitic ("LOPAR") fuel or optimized fuel assembly ("OFA") fuel. However, in the current period of transition, when mixed cores of LOPAR and OFA fuel are utilized at Turkey Point, the analysis results are slightly effected [sic] by the fact that the hydraulic resistance of OFA fuel is 4.5% higher than for LOPAR fuel. This ... results in approximately 10°F increase in PCT over the calculated 1972°F PCT for a homogenous core, which is well within the 2200°F criterion....

6. Licensee's Statement of Material Facts as to Which There is No Genuine Issue to be Heard adds the following:

> ECCS analysis has also been performed for a homogenous core with the previously approved evaluation model utilizing the Westinghouse Full Length Emergency Cooling Heat Transfer (FLECHT) correlation, resulting in an indicated peak clad temperature of 2130°F. A 10°F increase in temperature due to a mixed LOPAR and OFA core also results in a PCT less than the 2200°F limit....

7. Because of the extreme conditions prevailing in the event of a loss-of-coolant accident (LOCA), there are no mathematical models in existence which can accurately model the complex interactions between fuel, cladding, steam, and water which will take place during reflood. Classical mathematical analysis depends on assumptions of continuity which are clearly violated under accident conditions. There is no accepted body of mathematical knowledge which allows one to cope reliably with discontinuities such as those encountered in a LOCA-reflood sequence. Under such conditions, the mathematical models and computer codes developed to analyze conditions in the reactor core are, at best, vastly oversimplified conceptualizations based on the "average" behaviour of an "ideal" system (existing only in the mind) enjoying such desirable properties as uniformity, symmetry, and predictability — properties which do not necessarily correspond to reality. In short, accident analysis is an art with scientific underpinnings, but it is not an exact science. It predicts, not the precise behaviour of the system under study, but the "expected" or "probable" behaviour of the system under study. There is always a certain unquantifiable chance that the analysis itself is fundamentally wrong: wrong not just in its conclusions, but in its underlying assumptions.

8. The history of nuclear power is rife with examples of occurrences having major safety implications which were not predicted by prior mathematical analysis; for example:

(a) The 1966 accident at the Fermi nuclear reactor near Detroit was considerably worse than the "maximum credible accident" foreseen during the licensing process. Indeed, the accident involved the blockage of three fuel channels (with consequent overheating of the fuel) by a triangular piece of sheet metal which was nowhere indicated in any of the engineering specifications of the plant; the metal triangle had been added in the core area as an afterthought by a construction foreman.

The mathematical probability of the Fermi accident, prior to the onset of the accident, was "zero". No prior mathematical analysis could have taken into account the existence of the metal triangle, of which there was no record. In retrospect, of course, the mathematical probability of the Fermi accident can now be seen as quite high; indeed, very close to "one". Since the metal triangle was improperly welded, it was bound to come loose sooner or later; once loose, it was bound to obstruct the flow of coolant through the core; hence the accident.

An important lesson can be drawn from this example: mathematical probability is, to a large extent, a measure of our ignorance. The calculation of probability is affected by new information.

(b) The Three Mile Island accident offers many similar examples of events of "probability zero" which have actually occurred -- such as the simultaneous unavailability of all feedwater pumps, the onset of emergency cooling without a pipe break, and the initiation of an extremely serious accident because of the malfunction of a non-safety-related valve.

(c) Here in Canada, the sudden rupture of a pressure tube in the core of the Pickering Nuclear Generating Station (Unit 2) last summer was not only unforeseen, but was in direct contradiction to earlier mathematical analysis which had predicted a "leak-before-break" phenomenon. The tube suffered a longitudinal crack about one meter in length, intersected at right angles by a 270-degree circumferential crack, without any prior leakage.

(d) In the case of the Turkey Point Nuclear Generating Station, the degradation of the plant's steam generators to the point where replacement became necessary would not have been predicted by prior mathematical analysis, nor was the problem of pressure vessel embrittlement foreseen when the plant was first licensed.

In fact, nuclear technology is perhaps unique in becoming more and more complicated as time goes on, rather than more and more simple.

The number of new, fundamentally important, previously overlooked, still unresolved problems is greater in the nuclear field than in almost any other field of technology which has been commercially available for a comparable period of time.

9. Under the circumstances, it would be unwise to relax already existing safety margins solely on the basis of mathematical analysis using computer codes which predict that cladding failure will not occur in the event of LOCA, or that DNB will not occur otherwise. The following points should be borne in mind:

(a) The Licensee has just faced one major unanticipated problem by replacing the steam generators, and is now facing another major unanticipated problem: potential pressure vessel embrittlement. Until these problems have been fully resolved and the staff is fully familiar with the operating characteristics of the new core configuration, there should be no change in operating procedures which might result in shorter response time or more serious radioactive contamination in the event of a LOCA. Experience has shown that human error can often compound a relatively simple emergency, resulting in consequences much more serious than would have been anticipated.

(b) The mathematical analysis performed by the Licensee assumes a homogeneous full core of either LOPAR or OFA fuel. Since the transitional mixed core has not been studied as such, it would be wise to wait until after the transition is complete before translating the results of such analysis into licensing changes.

(c) The BART model used in the Licensee's analysis assumes (for purposes of mathematical convenience) that the system pressure is constant. In doing so, a number of phenomena which could significantly increase the cladding temperature have been excluded from consideration, including flow stagnation and flow oscillation. Moreover, the BART model does not encompass all possible expected flow patterns even if the system pressure is relatively constant.

(d) The BART model does not include a gap heat transfer model or a cladding swelling model, and such phenomena as embrittlement, blistering, hydriding, swelling, bowing, fission gas pressure, and possible reactions which might permanently affect heat transfer rates are not featured in the tests (involving electrically heated fuel arrays) that were used to ascertain the validity of the BART predictions. Such phenomena, by weakening certain localized parts of the fuel cladding, or by causing

undercooling in certain small regions, could materially affect the probability of cladding failure (since "a chain is only as strong as its weakest link"). To cope with such important problems simply by assigning a numerical "penalty" is little more than blind guesswork, reflecting the limitations of analysis.

(e) According to the analysis based on the FLECHT correlation, clad temperature could reach 2140°F: just 60°F short of the limit, which is 2200°F. Given the approximate nature of the analysis, it is purely a matter of judgment -- political, rather than technical in nature -- as to whether or not this is "too close for comfort", especially since it is known that the reflood will be slowed down by the increased hydraulic resistance of the mixed core and that the shut-off rods will take more time to drop into the core also.

(f) A computer analysis involving a great many calculations does not generally produce results which are perfectly accurate. Errors of various kinds creep into the results, some of which are easy to understand -- such as round-off error -- and others which are much more subtle. The Economic Council of Canada has reported that its CANDIDE model for the Canadian economy sometimes develops explosive cycles for no apparent reason, yielding results in which the error is very much larger than the answer. At present, there is no way of predicting in advance the maximum error that may obtain in any given computer analysis. This is an over-riding difficulty with all computer analysis involving large numbers of calculations.

(g) In its Safety Evaluation Report on the BART Code, referenced by Cecil O. Thomas of the Licensing Division on December 21 1983, we learn that "The BART code shows spikes in the calculated results of the heat transfer coefficients. The spikes are indicative of the discontinuous heat transfer regime transitions. However, the overall BART predictions are in good agreement with the heat transfer coefficient data." Of course, it is precisely where the "discontinuous heat transfer regime transitions" occur that cladding failure is likely to occur. Cladding failure generally begins as a local phenomenon, not necessarily as an "overall" phenomenon.

(h) From the same NRC document, we learn that "the analyses show that BART consistently overpredicts the average clad temperature." Again, it is not the average clad temperature that determined whether or not cladding failure will occur, but the localized clad temperature. In the vicinity of a discontinuity, an acceptably low average value does not translate into an acceptably low localized value.

(i) To prevent DNB, the NRC has employed the concept of a local DNB heat flux ratio, defined as the ratio of the critical heat flux (that would cause DNB at a particular core location) to the actual local heat flux: DNBR = CHF/AHF. If DNBR is less than or equal to 1, then DNB will occur. However, because of the

many uncertainties that exist, even if DNBR is calculated to be greater than 1, DNB may still occur (because the actual DNBR is not the same as the calculated DNBR). The DNB design basis is that there must be at least a 95 percent probability, with 95 percent confidence, that DNB will not occur provided that the calculated DNBR is greater than a certain specified limit. Thus, even if the calculated DNBR does not fall below the specified limit, there is still a chance that DNB may occur. Under the circumstance, if the reactor is operated in such a way that the fuel runs at a hotter temperature, even if the calculated DNBR is kept above the prescribed limit, the probability of DNB (resulting in fuel cladding failure) is materially increased.

(j) The Licensee argues that lowering the DNBR limit from 1.3 to 1.17, "in no way implies a reduction in the safety margin of a nuclear reactor" [Licensee's Motion for Summary Disposition of Intervenor's Contention (d)]. This statement is incorrect. By allowing the fuel to run at a hotter temperature, the reduction in DNBR limit does allow for a greater probability that DNB will occur. What the Licensee intends to say, perhaps, is that re-calculation indicated that the same margin of safety that was previously thought to exist can now be achieved at a higher operating temperature. Whatever the truth of this latter statement, it is undoubtedly true that running at a hotter temperature materially increases the probability of DNB, and therefore reduces the safety margin of the nuclear reactor.

(k) Given the limitations of mathematical analysis, all of these conclusions should be carefully tested against the actual operating experience of the plant. If the analysis is correct, then there should be no history of fuel failures and therefore no iodine releases.

10. Bo Lindell, Chairman of the International Commission for Radiological Protection, has commented on the oft-quoted dictum that all radiation doses should be kept "as low as reasonably achievable". He writes, "On the ALARA principle, doses near the dose limit would be rare, and the limit must now be seen as indicating the region of undisputed unacceptability rather than as guidance on what might still be acceptable" ["Basic Concepts and Assumptions Behind the New ICRP Recommendations", IAEA-SR-36/51, March 1979]. Since DNB or inadequate reflooding following a LOCA will result in releases of radioactivity, this is an appropriate philosophy to adopt in relation to reactor

safety as well. Contrast the Licensee's attitudinal stance, that running the fuel at a hotter temperature, provided it is within certain specified limits, "in no way implies a reduction in the safety margin of a nuclear reactor." Such a statement is not only technically incorrect, but inconsistent with a genuine concern for maintaining reactor risks "as low as reasonably achievable".

SWORN BEFORE ME

at Saint-Laurent, Province of Quebec

this 30th day of August 1984.

Adrien Leduc, Notary

MTRE ADRIEN LEDUC, Notary

AND I HAVE SIGNED.

Gordon Edwards

Gordon Edwards

EXHIBIT A

Professional Qualifications and Experience of

Gordon D. J. Edwards

My name is Gordon D. J. Edwards and my home address is 1300 Raimbault, Ville St. Laurent, Quebec, Canada, H4L 4R9. I am a professor of Mathematics and Science at Vanier College in Montreal, and I am also President of the Canadian Coalition for Nuclear Responsibility (Inc).

I graduated from the University of Toronto with a Bachelor of Science degree in Mathematics, Physics and Chemistry (gold medal in mathematics and physics) in June, 1961. As a Woodrow Wilson Fellow, I graduated from the University of Chicago with a Master of Science in Mathematics in June, 1962, and with a Master of Arts in English Language and Literature in June, 1964. After teaching university-level mathematics at the University of Western Ontario for several years, I graduated from Queen's University (Kingston, Ontario) with a Doctor of Philosophy degree in mathematics in June 1972. I conducted post-doctoral research in the Economics of Ocean Fisheries at the University of British Columbia in 1972-1973, and was the Assistant Director of a nation-wide study of the Mathematical Sciences in Canada for the Science Council of Canada in 1973-1974.

As an applied mathematician, I have been keenly interested in the strengths and weaknesses of mathematical modelling as it is applied to real-life problems. In 1977, I was a consultant to the Cluff Lake Board of Inquiry into Uranium Mining in Saskatchewan, where I cross-examined technical experts on such subjects as health effects of

radiation, radioactive waste disposal, and reactor safety. In 1977-1978, I was a consultant to the Ontario Royal Commission on Electric Power Planning, where I cross-examined experts in reactor safety from Ontario Hydro (the utility), Atomic Energy of Canada Limited (a research and development organization), and the Atomic Energy Control Board (the regulatory agency) over a period of several months. In 1979-1980, I was a consultant to the Select Committee on Ontario Hydro Affairs (a Committee of the Ontario Legislature) during a thirteen-week investigation into reactor safety following the Three Mile Island accident. I have also acted as a consultant on nuclear-related matters to such bodies as the Canadian Broadcasting Corporation, the National Film Board, the Science Council of Canada, the United Steelworkers of America, and several Canadian governmental bodies.

I have published several articles on nuclear power in Canada, with special reference to reactor safety and economics; in particular, "Cost Disadvantages of Expanding the Nuclear Power Industry" (Canadian Business Review, Spring 1982) and "Canada's Nuclear Dilemma" (Journal of Business Administration, Vol.13 1982). I have also prepared numerous unpublished reports involving technical critiques of various safety analyses for various clients.

United States of America
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
FLORIDA POWER & LIGHT COMPANY	)	Docket Nos. 50-250 OLA-2
	)	50-251 OLA-2
Turkey Point #3 & #4	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "INTERVENORS RESPONSE TO LICENSEES MOTION FOR SUMMARY DISPOSITION OF INTERVENORS' CONTENTIONS (b) and (d)," with attatched affidavits of Dr. Gordon Edwards and Joette Lorion, dated September 4, 1984, were served on the following by deposit in the United States mail, first class, postage prepaid and properly addressed on the date shown below.

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U.S. Nuclear Regulatory Commission
Washington, DC 20555

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Dated this fourth day of September 1984.