December 30, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Appeal Board

In the Matter of	)	
	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket Nos. 50-440 50-441
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF KEVIN W. HOLTZCLAW

State of California)
: ss:
County of Santa Clara)

Kevin W. Holtzclaw, being duly sworn, deposes and says as follows:

1. I, Kevin W. Holtzclaw, am a principal licensing engineer in the General Electric ("GE") Safety and Licensing Operation of the Nuclear Energy Business Operation. My business address is 175 Curtner Avenue, San Jose, CA 95125.

2. A statement of my professional qualifications is attached to this Affidavit as Exhibit "A". I have spent nearly 17 years as an engineer in the nuclear power industry, and have

worked for GE since 1969. I have spent over 12 years in GE Boiling Water Reactor ("BWR") fuel and core design and analysis. During this time I have performed numerous thermal and thermal-hydraulic evaluations for steady-state and anticipated transient conditions. As part of my work, I have evaluated thermal and thermal-hydraulic analyses of single loop operation ("SLO") of the recirculation system of BWRs.

3. In this Affidavit I respond to the five proposed contentions (Contentions "B-1" through "B-5") covering single loop operation of the recirculation system of the Perry Nuclear Power Plant, contained in the Motion to Reopen the Record and to Submit New Contentions (December 12, 1985) ("Motion") of Ohio Citizens For Responsible Energy ("OCRE"). For the reasons discussed in this Affidavit, the contentions are without technical merit. I am familiar with the technical information and references included in Appendix 15F (PNPP Single Loop Operation Analysis) to Amendment 22 to the Perry Final Safety Analysis Report ("FSAR"), dated November 20, 1985 ("Appendix 15F"), which OCRE's motion addresses. (Information identical to that contained in Appendix 15F had been earlier transmitted to NRC by letter from Cleveland Electric Illuminating Company dated October 28, 1985.) I am also familiar with other portions of the Perry FSAR which deal with issues OCRE has raised, as discussed in this Affidavit. I have personal knowledge of the matters set forth in this Affidavit and believe them to be true and correct.

4. Under the current Perry Nuclear Power Plant Unit 1 Technical Specifications ("Technical Specifications"), the Perry Plant is permitted to operate with a single reactor coolant system recirculation loop for a limited period of time (a maximum of 14 hours). See Technical Specifications, § 3/4.4.1.1.a. Under the current Technical Specifications, in the event that one recirculation loop is not in operation, immediate action must be taken to reduce power from the level the plant would be at in SLO (as high as 70%) to a level as high as 53%, within two hours. The Technical Specifications then permit the operators to remain in SLO at power levels up to 53% before putting the unit in hot shutdown within the next 12 hours. These provisions have been included in the Perry Technical Specifications since they were first docketed in mid-1984.

5. If approved by the NRC, Appendix 15.F of the FSAR would permit extended SLO of Perry up to 70% of rated thermal power. Seventy percent represents the approximate maximum power limit capability of Perry's BWR-6 reactor under SLO. GE has performed extensive generic BWR analyses, and Perry-specific analyses, which support the conclusions in Appendix 15F. These analyses demonstrate that Perry can safely operate with a single recirculation loop. See Appendix 15.F, pp. 15.F.1-1 - 15.F.1-2; 15.F.8-1 - 15.F.8-2 (References).

6. None of OCRE's proposed contentions on SLO provides a credible, technically-justified basis to question the analyses and conclusions set forth in Appendix 15F, which amply supports the conclusion that SLO is safe for Perry. Thus, none of OCRE's contentions raises a significant safety issue. I will address each of OCRE's contentions, and the arguments in support of the contentions, in the following paragraphs of this Affidavit.

CONTENTION B-1

7. Contention B-1 of OCRE's motion states:

> Applicants should analyze the progression and consequences of an anticipated transient without scram ("ATWS") initiated by the inadvertent startup of the idle recirculation loop when operating at 70% of rated thermal power with single loop operation. The analysis should demonstrate that this event will meet the safety criteria outlined in Section 15C.3 of the FSAR.

Motion at 2-3. The contention is without basis for the following reasons.

8. As stated in § 15.F.3.1 of Appendix 15F: "Inadvertent restart of the idle recirculation pump has been analyzed in the FSAR and is still applicable for single-loop operation." Appendix 15F, p. 15.F.3-2.

9. The idle recirculation loop startup transient is discussed in § 15.4.4 of the Perry FSAR. The transient analyzed in § 15.4.4 begins with reactor power at 54%. See Figure

15.4-1 (neutron flux curve at time zero). For this transient, the analysis demonstrates that scram is not required as a protective action. The transient response shows that no damage occurs to the fuel barrier and that the Minimum Critical Power Ratio ("MCPR") (the ratio of critical fuel bundle power to operating fuel bundle power, used to monitor the potential onset of boiling transition which could result in fuel damage) remains above (i.e., within) the applicable safety limit. See § 15.4.4.3.3. Therefore, ATWS analyses are irrelevant at power levels up to 54%.

10. If the idle recirculation loop startup transient is initiated at a thermal power level above 54%, conservative licensing basis calculations predict that Average Power Range Monitor ("APRM") flux scram would occur. FSAR § 15.4.4.2.3. Therefore, ATWS evaluations are relevant for this transient at power levels above 54%.

11. OCRE states at page 3 that the FSAR does not analyze an ATWS event initiated by the inadvertent startup of an idle recirculation loop. This ATWS event is not analyzed in the FSAR because it is less severe than events which are analyzed there. For the same reason, this event is not included in NEDE-25518 (referenced by OCRE at page 3), the Perry-specific analysis of ATWS events performed by GE.

12. The selection of ATWS events to be analyzed in NEDE-25518 and in the FSAR was based on prior GE generic

evaluations, including Volumes I and II of NEDO-24222 (Assessment of BWR Mitigation of ATWS (NUREG 0460 Alternate No. 3)) (1981). Volume II of NEDO-24222 concluded, for a BWR-6 such as Perry, that the idle recirculation loop startup event was less severe than the recirculation flow controller failure with increasing flow event, which in turn was less severe than the main steam isolation valve closure or turbine trip ATWS cases. NEDO-24222, vol. II, §§ 3.3.12, 3.3.13. Perry-specific analyses of these four transients show the same relative severity based on neutron flux and heat flux predictions, with the idle recirculation loop startup event (FSAR § 15.4.4) and recirculation flow controller failure (FSAR § 15.4.5) being much less severe than the turbine trip (FSAR § 15.2.3) and main steam isolation valve events (FSAR § 15.2.4).

13. The idle recirculation loop startup event discussed in NEDO-24222 and FSAR § 15.4.4 was based on power levels of less than 70%. However, the event at a 70% power level is less severe than at the levels assumed in NEDO-24222 and FSAR § 15.4.4. The reason why the event starting at a higher power level is less severe is that the power rise from beginning to peak is less for the 70% case. The relative core flow change that would be caused by the idle pump startup at 70% power is smaller because the reactor is already operating at a higher core flow than it would be at lower power levels. The evaluations in NEDO-24222 are bounding (and were chosen because they

were bounding), and therefore the idle recirculation loop startup event at 70% power need not be analyzed for ATWS.

14. NEDO-10349, referenced by OCRE at page 3 of the Motion, is a 1971 analysis based on earlier GE BWR models. However, the conclusion in NEDO-10349, that the idle recirculation loop startup transient is less severe than recirculation flow controller malfunction - increasing flow ATWS, which OCRE cites in its motion, is entirely consistent with GE's later transient analyses for BWR-6 models such as Perry, as discussed above.

15. For these reasons, further ATWS analyses based on recirculation loop startup are not required to demonstrate the safety of SLO at Perry.

CONTENTION B-2

16. Contention B-2 of OCRE's motion states:

> Applicants have not demonstrated that the seizure of the operating recirculation pump when operating up to 70% of rated thermal power with a single loop will not exceed fuel safety limits, assuming scram functions, and that ATWS initiated by this event will meet the safety criteria of FSAR Section 15C.3.

Motion at 3-4. The contention is without basis for the following reasons.

17. As indicated in § 15.F.3.1 of Appendix 15F: "The one recirculation pump seizure accident has been reviewed for single loop operation. Results show that this accident poses no threats to thermal [i.e., fuel safety] limits." As part of

this review, GE specifically analyzed the recirculation pump seizure accident for thermal power levels up to 70% power in SLO at Perry. The results of GE's analysis show that no safety limits were exceeded.

18. OCRE's first argument is that steam blanketing of the fuel rods would occur if a recirculation pump seizure accident occurred in SLO. GE's analysis shows that steam blanketing of the fuel rods (i.e., formation of a steam layer around the fuel rods sufficient to prevent liquid from cooling the fuel cladding) would not occur in the unlikely event of recirculation pump seizure. I have reviewed page 32 of Richard Webb's "The Accident Hazards of Nuclear Power Plants" (1976), cited by OCRE at page 4 of its motion. The cited portion of Mr. Webb's book does not refer in any way to SLO. Based on my extensive experience in fuel and core design and analysis, I know of no technical basis with respect to BWR's for Mr. Webb's unsupported assertion that steam blanketing could occur and threaten fuel rod breakup following a hypothetical instantaneous coolant pump seizure.

19. OCRE's second argument (pages 4-5) is that a pump seizure in SLO would put the reactor into natural circulation, that neutron flux oscillations might then occur, and that an ATWS might then take place which would prevent operators from inserting control rods to suppress the oscillations. GE's analysis of the pump seizure accident during SLO for Perry,

discussed in ¶ 17, demonstrated that no scram is required following such an accident. The scram at 3.4 seconds into the pump seizure accident (cited by OCRE at p. 4) only occurs during normal two loop operation and would not occur in SLO. Thus, contrary to OCRE's contention, ATWS is not a concern for the SLO pump seizure accident postulated by OCRE.

20. Aside from the fact that ATWS is irrelevant to an SLO pump seizure accident, OCRE's argument fails to recognize the difference between the scram function and manual control rod insertion capability. The neutron flux oscillations postulated by OCRE are suppressed by manual control rod insertion, not by the scram function. There is no technical justification or basis to postulate loss of all control rod insertion capability in addition to postulating the highly improbable pump seizure accident, as OCRE's contention would require. Control rod insertion capability is assured at Perry through a number of plant systems which are designed to meet single-failure, safety-grade requirements. Thus, even if a pump seizure accident were to occur, the control rods could be inserted which would suppress any resulting oscillations.

21. GE analyses (including recommended operational guidelines adopted for Perry) also demonstrate that potential neutron flux oscillations would not result in Perry fuel design limits being exceeded in SLO. NEDE-22277, "Compliance of General Electric Boiling Water Reactor Fuel Designs to Licensing

Criteria" (October 1984) and App. 15F, § 15.F.4. The NRC accepted the GE analysis for referencing on April 24, 1985 (subsequent to the March 1984 Board Notification relied upon by OCRE) (see App. 15F, ref. 15.F.8-6).

22. For these reasons, further analyses of pump seizure during SLO for Perry are not necessary to assure that the safety criteria of FSAR Section 15C.3 would be met.

CONTENTION B-3

23. Contention B-3 of OCRE's motion states:

> Applicants have not demonstrated that the traversing incore probe ("TIP") noise uncertainty values reported in the FSAR Section 15.F.2.2 are applicable to single loop operation up to 70% of rated thermal power; consequently, the minimum critical power ratio ("MCPR") may not be determined in a conservative fashion.

Motion at 5. The contention is without basis for the following reasons.

24. The TIP noise value, referenced in this contention, is a random variation of the measured reading of the reactor local power level. TIP reading uncertainty values are used in determining the MCPR safety limit. See FSAR §§ 15.F.2. As reflected in FSAR § 15.F.2.2, rather than relying on theoretically derived TIP noise values, GE was able to use actual noise values from an SLO test performed at an operating BWR, to determine the TIP noise uncertainty for the Perry SLO MCPR fuel cladding integrity safety limit analysis.

25. There would be no significant difference in TIP noise seen at 59.3% thermal power, as in the test referenced in § 15.F.2.2, and TIP noise values that would be expected for SLO up to 70% of rated power for SLO at Perry. This is because the major contributors to the TIP noise, as determined in previous GE generic BWR analyses, are geometric mislocation of the TIP detector and neighboring fuel channels with respect to their nominal design positions, and the random neutron, electronic and boiling noise in the reactor. See NEDO-20340, GE Licensing Topical Report, "Process Computer Performance Evaluation Accuracy" (June 1974) (referenced in FSAR § 4.3.5). In fact, the percent uncertainty of TIP noise will decrease as power increases.

26. For these reasons, the TIP noise uncertainty given in FSAR § 15.F.2.2, and the MCPR safety limit analysis in § 15.F.2.2, are applicable and conservative for SLO at Perry up to 70% power. Therefore, contrary to OCRE's contention, no further analysis of TIP noise uncertainty and the resulting MCPR compliance is necessary or appropriate.

CONTENTION B-4

27. Contention B-4 of OCRE's motion states:

> Applicants' Technical Specifications for single loop operation up to 70% of rated thermal power should include limits on the core plate pressure drop.

Motion at 5. The contention is without basis for the following reasons.

28. The core plate pressure drop measures differential pressure between the core inlet and outlet flow conditions. The general information that can be obtained by measuring core plate pressure drop is provided in a more direct and useful form by measuring core power and flow. Appropriate proposed Perry Technical Specification limits for SLO, which will include core power and flow, will be submitted to the NRC based on the results of start-up tests which are part of the power ascension program. For this reason, no core plate pressure drop limits are needed in the Perry Technical Specifications for SLO.

29. The current technical specifications for the Cooper Nuclear Station, the plant referenced by OCRE at pages 5-6 of its motion, do include a limitation on fluctuation in the core plate differential pressure. This fluctuation involves small variations in the measured pressure. Since the Perry Technical Specifications will adequately monitor and control core power and flow during SLO through appropriate Technical Specification

limits, there is no need to include separate limits on core plate pressure drop fluctations.

30. Contrary to OCRE's assertions at pages 5-6 of the Motion, Technical Specification limits on core plate pressure drop (or on core plate pressure drop fluctation), would not permit better regulation of core flow. As noted in prior GE analyses, approved by the NRC, there is negligible flow assymetry (flux tilt) during SLO. See NEDO-10722A, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1" (August 1976). Since power and flow are not tilted due to SLO, Technical Specification limits on core plate pressure drop, or on core plate pressure drop fluctuation, would not lead to less variable within-core coolant flow, or yield more even cross-core power, as OCRE asserts at page 5 of its motion.

31. For these reasons, contrary to OCRE's contention, there is no need to include in the Perry Technical Specifications SLO limits on core plate pressure drop (or on core plate pressure drop fluctuation), and such limits have not been included in the Technical Specifications.

CONTENTION B-5

32. Contention B-5 of OCRE's motion states:

> Applicants have not demonstrated that single loop operation up to 70% of rated thermal power will not aggravate the strong variability in flow rate along the fuel channel seen in fast BWR transients, or that this phenomenon has been conservatively accounted for in analyses of fast transients.

Motion at 6. The contention is without basis for the following reasons.

33. In the article on "Critical Power Ratio in BWR Transient Analyses," cited by OCRE at page 6 of its motion, the authors correctly note that flow rate may vary strongly along the fuel channel during a fast transient. The article makes no distinction between fast transients during SLO and normal two loop operation, and does not suggest that this fuel channel flow rate phenomenon would be more severe during SLO fast transients. I know of no technical basis indicating that this phenomenon would be aggravated in a fast transient initiated during SLO operation.

34. The authors of the article cited by OCRE applied GE's methodology for determining Critical Power Ratio ("CPR") (the same methodology used in the FSAR Appendix 15F analysis), as well as an iterative method which resulted in a lower (more conservative) CPR. The article concludes that for fast transients, "the values of CPR calculated by both methods remain

within the required safety limits." The article thus does not indicate that GE's methodology is inappropriate for determining CPR in fast BWR transients. The NRC has reviewed and approved GE's CPR methodology, which is described in the GE licensing topical report listed as Reference 15.F.8-1, FSAR § 15.F.8, page 15.F.8.1.

35. Fast transients such as feedwater controller failure (maximum demand) and generator load rejection with bypass failure, were specifically analyzed by GE in support of FSAR § 15.F.3. See FSAR §§ 15.F.3.1.1, 15.F.3.1.2. These FSAR references demonstrate that MCPR for both cases remains well within safety limits.

36. There is no technical basis for OCRE's statement at page 6 of its motion, that SLO at Perry would involve "non-uniform flow rates throughout the core." As noted in ¶ 30 above, and in the NRC-approved topical report referenced therein, there would be negligible flow assymetry during SLO.

37. OCRE also suggests at page 6 of the Motion that the increased flow measurement uncertainty during SLO, as discussed in FSAR § 15.F.2, might aggravate variability in flow rate along the fuel channel in fast BWR transients. OCRE suggests no technical basis for this statement, and there is none. Increased flow measurement uncertainty during SLO is explicitly accounted for in the Appendix 15F analysis establishing core and fuel limits, as discussed in FSAR § 15.F.2. Core flow

measurement uncertainty is thus factored into GE's calculation of the CPR used in evaluating all of the transients considered in Appendix 15F. Increased core flow measurement uncertainty cannot affect, or "aggravate," channel flow variability during fast transients, as suggested by OCRE. Core flow measurement uncertainty is related only to initial core flow and not to channel flow variability during the transient.

38. For these reasons, there is no basis for OCRE's suggestion that SLO at Perry might aggravate the variability in flow rate along the fuel channel in fast BWR transients, or that Applicants have failed to conservatively account for this phenomenon in their SLO analysis of fast transients.

39. For the reasons stated above, Contentions B-1 through B-5 in OCRE's motion are without any credible technical basis. The contentions fail to call into question the adequacy of Applicants SLO analysis contained in Appendix 15F, and fail to raise any safety concerns.

Executed at San Jose, California, this 30th day of December, 1985.

Kevin W. Holtzclaw

Kevin W. Holtzclaw

Subscribed and sworn before me this 30th day of December, 1985.

OFFICIAL SEAL
PAULA F HUSSEY
NOTARY PUBLIC - CALIFORNIA
SANTA CLARA COUNTY
My comm. expires JAN 13, 1989

Paula F. Hussey

NOTARY PUBLIC, STATE OF CALIFORNIA

EXHIBIT A

STATEMENT OF PROFESSIONAL QUALIFICATIONS
KEVIN W. HOLTZCLAW

Education: B.S. Mechanical Engineering (Nuclear Option), San Jose State University, M.S. Mechanical Engineering, University of California, Berkeley, General Electric Advanced Courses in Engineering

Experience:

- March 1982 to Present: Principal Licensing Engineer, Program Manager of the GE Severe Accident Program
- February 1980 to March 1982: Senior Licensing Engineer, BWR Systems Licensing (GE)
- June 1974 to February 1980: Technical Leader, Fuel Applications & Thermal Design (GE)
- January 1971 to June 1974: Engineer, Fuel Applications and Thermal Design (GE)
- July 1969 to January 1971: Program Engineer Fuel Performance & Applications (GE)
- June 1968 to January 1971: Engineer - Nuclear Power Department (San Francisco Bay Naval Shipyard)

Licensing Experience:

Approximately 17 years engineering experience in the nuclear plant power industry. Since 1980, concentration on BWR licensing issues. As a senior licensing engineer through 1982, responsible for defining and planning programs related to NRC degraded core rulemaking. Responsible for the safety and licensing program management of the Limerick Probabilistic Risk Analysis. GE representative on AIF Industry Degraded Core Rulemaking Technical Advisory Group.

Since March 1982, GE Program Manager of the GE Severe Accident Program. This has entailed managing the BWR/6 standard plant Probabilistic Risk Assessment and Severe Accident submittals relating to evaluations beyond current design bases. Continued as the GE representative on the Industry Degraded Core Rulemaking (IDCOR) Technical Advisory Group. Responsible engineer in the GE Safety and Licensing organization for the GE Fission Product Retention Program and Severe Accident Source Terms and for programs relating to hydrogen generation and control. Numerous presentations to domestic and foreign regulatory groups and nuclear societies.

Additional Work Experience

Engineer and technical leader in General Electric's Fuel Design Department from 1969 to 1980 responsible for performing Reload and Initial Core Fuel Thermal and Thermal-Hydraulic fuel design and safety analyses. Principal responsibilities in development of thermal analysis methods, the design and licensing of 8x8 fuel and extended exposure fuel designs, and in defining acceptance criteria for fuel thermal-mechanical fuel integrity properties and capabilities. Mechanical design engineer in the nuclear power department of the San Francisco Bay Naval Shipyard.

December 30, 1985

UNITED STATES OF AMERICA
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Before the Atomic Safety and Licensing Appeal Board

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	)	
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket Nos. 50-440 50-441
	)	
(Perry Nuclear Power Plant, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

This is to certify that copies of the foregoing APPLICANTS' ANSWER TO OCRE MOTION TO REOPEN THE RECORD AND TO SUBMIT NEW CONTENTIONS were served by deposit in the United States Mail, first class, postage prepaid, this 30th day of December 1985, to all those on the attached Service List.

Jay E. Silberg

Jay E. Silberg

DATED: December 30, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket Nos. 50-440 50-441

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