UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	))))))))	Docket No. 50-440 OL 50-441 OL

NRC STAFF ANSWERS TO OCRE NINTH SET OF INTERROGATORIES TO NRC STAFF

On January 31, 1983 Ohio Citizens for Responsible Energy (OCRE) filed its "Ninth Set of Interrogatories to the NRC Staff" and requested that the Licensing Board direct the Staff to provide answers. By letter to the Licensing Board dated February 17, 1983 NRC Staff counsel advised the Licensing Board that the Staff would voluntarily answer OCRE's interrogatories and intended to serve the answers by March 1, 1983.

The Staff's answers (with the affidavits and statements of the professional qualifications of their preparers) to OCRE's "Ninth Set of Interrogatories to the NRC Staff" are attached.

Respectfully submitted,

James M. Cutchin IV
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 1st day of March, 1983

DS07

ANSWERS TO OCRE'S NINTH
SET OF INTERROGATORIES TO NRC STAFF

ISSUE #13

INTERROGATORY NO. 9-1

Why has the Staff reopened the issue of turbine missile hazards at PNPP at the OL stage when this issue was considered resolved at the CP stage?

ANSWER

The results of turbine inspections at operating nuclear facilities over the past several years indicate that cracking to various degrees has occurred at the inner radius of turbine disks, particularly those of Westinghouse design. Within this time period there has actually been a Westinghouse turbine disk failure at one facility owned by the Yankee Atomic Electric Company. Recent inspections of General Electric turbines have also resulted in the identification of disk keyway cracks.

In view of this operating experience and in the interest of maintaining NRC safety objectives, we now are emphasizing the turbine missile generation probability (i.e., turbine system integrity) in our reviews of the turbine missile issue and eliminating the need for elaborate and generally ambiguous analyses of strike and damage probabilities given an assumed turbine failure rate.

INTERROGATORY NO. 9-2

Has the Staff made any interim or preliminary findings as to the risk of turbine missile hazards at PNPP? If so, produce same.

ANSWER

Yes, the Staff has concluded that the probability of unacceptable damage to safety-related structures, systems, and components due to turbine missiles is acceptably low (i.e., less than 10^{-7} per year) provided that the total turbine missile generation probability is kept below 10^{-5} per reactor year, throughout the life of the plant, by an acceptable maintenance program.

The Staff is not aware of any turbine rotor rupture due to crack propagation (i.e., brittle fracture) that has occurred within three years of startup. Moreover, no cracks with depths greater than one-half the critical crack depth calculated for that wheel have been observed in a General Electric turbine wheel within three years of startup. For these reasons, the Staff intends to allow the Applicants up to three years from initiation of power output to propose a turbine maintenance program (inspection and testing procedures and schedules) and obtain NRC Staff approval of the program.

INTERROGATORY NO. 9-3

Why did the Staff at the CP stage consider the Perry design acceptable with regard to turbine missile hazards when the Staff's estimates of combined strike probability (1.4×10^{-2}/year) and overall probability for damage (5.5×10^{-7}/year) exceeded the values given in Regulatory Guide. 1.115 (1×10^{-3}/year and 1×10^{-7}/year, respectively)? (The Staff's estimates were stated in Supplement 5 of the SER-CP.)

ANSWER

As discussed in Section 2.2.3, "Evaluation of Potential Accidents," of the SRP (NUREG-0800) an estimated probability of unacceptable damage of 10^{-6} per year is acceptable if sufficiently conservative procedures were used in making the calculation. At the time of the CP stage review, the Staff apparently considered the conservatisms presented in Supplement 5 of the SER-CP to be sufficient for the estimated overall damage probability of 5.5×10^{-7} per year to be acceptable.

INTERROGATORY NO. 9-4

Has the Staff taken any regulatory position concerning the preferred orientation of turbine generators (i.e., tangential vs. radial) with respect to safety-related structures? If so, produce the document expressing this position.

ANSWER

Yes, the Staff's position is that turbine generator orientation is a contributing, but not a dominating, safety factor with respect to potential turbine missile damage to safety-related structures, systems and components. It is advantageous to orient the turbine generator favorably, i.e., to orient it such that all safety-related structures, systems, and components are outside the low trajectory turbine missile strike zone. It is the Staff's view that the choice of a plant design with a favorable orientation decreases by about an order of magnitude the probability of unacceptable damage to safety-related structures, systems, and components compared to a design with an unfavorable turbine orientation.

The dominating safety factor with regard to potential turbine missile damage is the prevention of missile producing turbine failures by adequate design, manufacture, and maintenance procedures. In general, unfavorably oriented turbines must be inspected more frequently than favorably oriented ones to assure the same probability of unacceptable damage to safety-related structures, systems, and components due to turbine missiles.

This position will be documented in a future supplement to the Perry SER (NUREG-0887).

INTERROGATORY NO. 9-5

Does the Staff have any preferred methods for calculating probabilities of turbine missle damage? If so, describe any such methods in detail and explain why they are preferred.

ANSWER

Yes, the Staff has a preferred method for calculating probabilities of turbine missile damage. The probability of unacceptable damage due to turbine missiles (P_4) is preferably expressed as the product of (a) the probability of turbine failure resulting in the ejection of turbine disk (or internal structure) fragments through the turbine casing (P_1), (b) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components (P_2), and (c) the probability of struck structures, systems, or components failing to perform their safety function (P_3).

According to the current NRC guidelines stated in Section 2.2.3 of the SRP (NUREG-0800) and Regulatory Guide 1.115, the probability of unacceptable damage from turbine missiles should be less than or equal to one chance in ten million per year for an individual plant, i.e., P_4 should be less than or equal to 10^{-7} per year.

Based on estimates for a variety of plants the Staff has concluded that if a turbine missile is generated the probability of unacceptable damage to safety-related structures, systems and components would be significant, i.e., approximately 10^{-3} per year or 10^{-2} per year depending on whether turbine orientation is favorable or unfavorable. For this reason and because of the experience with the cracking of turbine disks (see the answers to Interrogatories Nos. 9-1 and 9-2), the Staff has shifted its review emphasis to the prevention of missile-generating turbine failures.

This shift of emphasis necessitates that nuclear steam turbine generator manufacturers develop and implement volumetric (ultrasonic) examination techniques suitable for inservice inspection of turbine disks and shaft, and prepare reports for NRC Staff review which describe the methods for determining turbine missile generation probabilities. These methods are to relate disk design, materials properties, and inservice volumetric inspection interval to the design speed missile generation probability, and to relate overspeed protection system characteristics, and stop and control valve design and inservice test interval to the destructive overspeed generation probability.

Westinghouse and General Electric are in the process of establishing models and methods for calculating turbine missile generation probabilities for their respective turbine generator systems and of supplying on a plant specific basis utilities with their turbines the missile generation probability as a function of rotor inspection and valve test intervals. These data will be used by utilities to select rotor inspection and valve test intervals which keep the missile generation probability below 10^{-5} per year for unfavorably oriented turbines and 10^{-4} per year for favorably oriented turbines.

INTERROGATORY NO. 9-6

It is stated in "A Reassessment of Turbine-Generator Failure Probability" by S.H. Bush, Nuclear Safety, Vol. 19, No. 16, Nov.-Dec. 1978 at 681 that any reassessment of P_3, the probability of significant damage to components and structures from a missile strike, must await the completion of jet-sled missile tests sponsored by the Electric Power Research Institute (EPRI). If such tests are now completed, and if the Staff possesses the results of such tests, produce said results, and explain how these results affect probability calculations for determining turbine missile hazards.

ANSWER

Results of completed jet-sled missile tests that were sponsored by EPRI and the use of data from those tests in assessing turbine missile risks are addressed in "Turbine Missile Risk Methodology and Computer Code" and "Concrete Impact Prediction Techniques" by Lawrence A. Twisdale, et al. These papers were presented at an "EPRI Seminar on Turbine Missile Effects in Nuclear Power Plants" in Palo Alto, California on October 25-26, 1982 and are being made available in both the NRC PDR and the Perry LPDR for inspection and copying.

INTERROGATORY 9-7

Explain in detail how the probability estimates given in Supplement 5 of the Perry SER-CP were calculated.

ANSWER

The Staff member currently responsible for assessing turbine missile damage probabilities at PNPP is not familiar with the details of how the probability estimates given in Supplement 5 of the Perry SER-CP were made. However, he believes that the general procedure outlined in Section 3.5.1.3 of the SRP (NUREG-0800) was followed. In any case, the Staff is not making use of the estimates presented in Supplement 5 of the Perry SER-CP in their OL stage review, but rather, for the reasons stated in the Staff's answer to Interrogatory No. 9-1, is conducting the OL stage review of the turbine missile issue for Perry by the procedure outlined in the Staff Answer to Interrogatory No. 9-5.

INTERROGATORY NO. 9-8

Explain the bases for the Staff's use of the probability values (1×10^{-3}/year for combined strike probability and 1×10^{-7}/year for overall turbine missile hazards) given in Regulatory Guide 1.115.

ANSWER

The basis for the combined strike probability of 10^{-3} per year, assuming an overall turbine missile hazards probability of 10^{-7} per year, is discussed in Section 3.5.1.3, "Turbine Missiles," of the SRP (NUREG-0800) and in Regulatory Guide 1.115. For a discussion of an acceptably low probability for events such as unacceptable turbine missile damage to safety-related structures, systems and components, see Section 2.2.3, "Evaluation of Potential Accidents" of the SRP (NUREG-0800). An event with a probability of occurrence of 10^{-7} per year is viewed by the NRC Staff as "incredible."

INTERROGATORY NO. 9-9

Describe in detail each and every portion, statement, or methodology in GAI Report No. 1848, "An Analysis of Low Trajectory Turbine Missile Hazards, Perry Nuclear Power Plant, Units 1 and 2," October 1976, which the Staff finds unacceptable or of questionable basis, and indicate why.

ANSWER

While the Staff is aware of some of the procedures presented in GAI Report No. 1848, "An Analysis of Low Trajectory Turbine Missile Hazards, Perry Nuclear Power Plant, Units 1 and 2," October 1976, the Staff has not "reviewed" this report, and has no plans to do so.

INTERROGATORY NO. 9-10

If Applicants have submitted any other, additional documentation concerning turbine missile hazards, identify such documentation and describe each and every portion of any such documentation which the Staff finds unacceptable or of questionable basis, and explain why.

ANSWER

No additional documentation concerning turbine missile hazards has been submitted by the Applicant.

ISSUE #14

INTERROGATORY NO. 9-11

Explain why the Staff no longer requires in-core thermocouples in BWRs, as indicated by Section 4.4.7 of NUREG-0887, the Perry SER.

ANSWER

The Staff presently does not "require" installation of in-core thermocouples in BWR's, because, as is discussed in Section 4.4.7 of the Perry SER, in-core thermocouples alone are not well responding unambiguous indicators of inadequate core cooling under some conditions. Instead of "requiring" installation of in-core thermocouples, the Staff is providing BWR applicants an opportunity to demonstrate that other available means of detecting inadequate core cooling are adequate.

As described in the answer to Interrogatory No. 9-14, the ACRS recommended that the in-core thermocouple requirement be reevaluated by the Staff. Additionally, the BWR Owners Group (BWROG) contracted a study (see answer to Interrogatory No. 9-15) which concluded that the effectiveness of in-core thermocouples as an inadequate core cooling indicator is very limited and led the BWROG to recommend to the Staff that in-core thermocouples not be used to detect inadequate core cooling.

The Staff in reviewing the BWROG recommendation questioned the reliability of existing water level instrumentation as the sole indication of inadequate core cooling. The Staff requested that a further study be performed by the BWROG to evaluate the need for upgrading existing water level instruments to make them more reliable inadequate core cooling detectors and that the BWROG consider what other instrumentation (including in-core thermocouples) might be needed in the BWR plant monitoring system. The Staff's position on conformance of the Perry plant with TMI Action Plan Item II.F.2 of NUREG-0737 was stated as follows in the Perry SER:

> As a result of these [BWROG/NRC] meetings, agreements have been reached to broaden the issue from the specific requirements for in-core thermocouples to that of monitoring inadequate core cooling. The BWR Owners Group has agreed to actively participate in the analysis of inadequate core cooling requirements and will be submitting a final report for the staff's review in July 1982.

In Section 4.4.7.2 of the Perry SER the Staff further stated:

> [T]he operating license of Perry will be conditioned for the submittal of this [the BWROG] report by July 1982 and to require conformance with any Item II.F.2 requirements which result from the Staff's evaluation of that report.

The BWROG submitted to the Staff in August 1982 a report entitled, "Review of BWR Reactor Water Level Measurement Systems," SLI-8211," dated July 1982, which includes the BWROG's evaluation of existing water level instruments and recommendations for their improvement. In December 1982 the BWROG submitted a second report entitled, "Inadequate Core Cooling Detection in BWR's," SLI-8218, dated November 1982, which presents evaluation results of additional instrumentation as diverse indicators of inadequate core cooling with recommendations regarding the need for such additional instrumentation (including in-core thermocouples) for

BWR plant monitoring systems. At the Staff's request, the Applicants also submitted a plant specific evaluation (in a letter to NRC dated January 14, 1983) addressing the applicability of the BWROG findings (in SLI-8211 and SLI-8218) to Perry.

The Staff expects to complete its review of the BWROG reports and the Applicant's submittal in late Summer 1983. The results of the Staff's review will be reported in a future supplement to the Perry SER.

INTERROGATORY NO. 9-12

What types of instrumentation does the Staff consider acceptable for the detection of inadequate core cooling in BWRs?

ANSWER

The Staff considers instruments that measure coolant temperature, level or inventory to be potentially acceptable for detecting inadequate core cooling in BWR's.

The first indication of an approach to an inadequate core cooling condition in a BWR is a drop in water level which threatens core uncovery. Therefore, the Staff considers existing water level instrumentation to be acceptable for plant interim operation. The Staff is evaluating whether improvements recommended in BWROG Report SLI-8211 are needed to upgrade the reliability of water level measurement as the primary indicator of inadequate core cooling. Other types of instrumentation, to be acceptable to the Staff for the detection of inadequate core cooling, must provide an unambiguous indication to the extent that any ambiquity can be readily recognized and correctly interpreted by the plant operator(s). Functional and design requirements for such other instrument are provided in TMI Action Plan Item II.F.2 of NUREG-0737.

INTERROGATORY NO. 9-13

Did the Staff at any time have any specific requirements for the placement of in-core thermocouples, for thermocouple characteristics or any other criteria? If so, produce same.

ANSWER

Yes, the design characteristics and other criteria for BWR in-core thermocouples are specified in Revision 2 of Regulatory Guide 1.97 (December 1980) entitled, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environ Conditions During and Following an Accident." Specifically, the monitoring temperature range of thermocouples used as a diverse indicator of water level should extend from 300°F to 2300°F. In addition, the Staff recommends the placement of four thermocouples in each core quadrant, with at least one thermocouple per quadrant operable during plant operation.

INTERROGATORY NO. 9-14

SECY-81-582 states that the ACRS supported the use of in-core thermocouples in BWRs. Does the ACRS support the Staff's new policy of not requiring thermocouples? Provide documentation of the ACRS position.

ANSWER

The ACRS position on the use of in-core thermocouples for detection of inadequate core cooling is stated on page 2 of ACRS Report No. 0938, dated August 11, 1981. In pertinent part it is stated:

> The NRC Staff proposed to require the installation of core thermocouples in the Susquehanna Station as specified by Regulatory Guide 1.97, Revision 2, . . . Applicant has not yet agreed to this requirement. We supported the use of thermocouples in BWR's in our letter of November 10, 1980 to the NRC Executive Director but called attention to the need for further study to determine the appropriate vertical location of such thermocouples. Since most of the information of interest from thermocouples may be obtainable from a small number of thermocouples placed in a more accessible location, we recommend that this requirement be reevaluated.

In response to the ACRS recommendation, and the study findings contained in the BWROG reports under Staff review, the Staff revised the position "requiring" in-core thermocouples as described in the answer to Interrogatory No. 9-11.

INTERROGATORY NO. 9-15

(a) Has the Staff reviewed the document "Thermal Analysis of In-Core Thermocouples in Boiling Water Reactors" prepared for the BWR Owners Group by S. Levy, Inc. (November 1981)?

(b) If so, does the Staff agree with the calculations, arguments, and conclusions presented therein?

(c) Describe every calculation, statement, argument, or conclusion in the document with which the Staff disagrees, or finds unacceptable or of questionable basis, and explain why.

ANSWER

(a) The document "Thermal Analysis of In-Core Thermocouples in Boiling Water Reactors," prepared for the BWR Owners Group by S. Levy, Inc. is included in the BWROG Report SLI-8218 as Appendix B dated November 1982. The Staff has read the document but has not formally reviewed it. The Staff's report of its evaluation of BWROG Report SLI-8218 will address the document that is included as Appendix B. The Staff's report is scheduled to be issued in late Summer 1983.

(b) & (c) As stated above, the Staff has not completed its evaluation of the findings presented in BWROG Report SLI-8218. The document that is included as Appendix B to SLI-8218, as discussed in the BWROG report, concludes that the calculated response delay time of in-core thermocouples is at least 10 minutes (i.e., the thermocouples will not respond for at least 10 minutes after core uncovery in a small break LOCA) and that in-core thermocouples will provide ambiguous information to plant operators during the delayed response period, e.g., in-core thermocouples would indicate to the operator that the plant is not in trouble while the existing water level instrumentation would indicate that the plant is in trouble. The assumed heat-up rate used in the BWROG analysis is consistent with a decay power of 2% of initial power, which corresponds to a time period of 700 seconds from reactor shutdown to beginning of core uncovery. The Staff agrees that this is a reasonable assumption to represent a power level for core uncovery resulting from a typical small break LOCA. The Staff also agrees that in-core thermocouples alone do not provide an unambiguous indication of a core uncovery condition.

INTERROGATORY NO. 9-16

(a) Has the Staff reviewed the document "General Electric Evaluation of the Need for BWR Core Thermocouples" dated November 16, 1981?

(b) If so, does the Staff agree with the calculations, arguments, and conclusions presented therein?

(c) Describe every calculation, statement, argument, or conclusion in the document with which the Staff disagrees, or finds unacceptable or of questionable basis, and explain why.

ANSWER

(a) The Staff has read the document but has not formally reviewed it.

(b) As to the conclusion that existing BWR water level instrumentation is adequate to indicate the approach to inadequate core cooling and in-core thermocouples are not needed, see the answers to Interrogatories Nos. 9-11 and 9-12.

(c) See the answers to Interrogatories Nos. 9-11 and 9-12.

ISSUE #15

INTERROGATORY NO. 9-17

Has the Staff formulated any regulatory policy, statement, criteria, plans, or other position concerning steam erosion and its effects, causes, prevention, detection, or mitigation? If so, produce same.

ANSWER

No, the Staff has not defined any regulatory policy, statement, criteria or plans concerning steam erosion and its effects, causes, prevention, detection, or mitigation. None of the components known to the Staff to have experienced steam erosion, as described in I&E Information Notice 82-22, is classified as ASME Code Class 1, 2, or 3. Therefore, these components are not safety-related, because they are not part of reactor coolant pressure boundary, nor are they relied upon to safely shut down the reactor or to mitigate the consequences of postulated accidents. Thus, these components are not included in inservice inspection programs based on the requirements of paragraph 50.55a(g) of 10 CFR Part 50. However, a few of the main steam isolation valves (MSIV's) identified in IE Information Notice 82-23 did experience steam erosion, and MSIV's are safety-related (ASME Code Class 1). Also, see the Staff's answer to Interrogatory No. 9-20 (a) and (b).

INTERROGATORY NO. 9-18

Has the Staff (or anyone to its knowledge or on its behalf) conducted any research or studies in an attempt to determine the causes, effects, and means of prevention, detection, and mitigation of steam erosion? If so, produce any such research or studies.

ANSWER

Steam erosion of pressure boundary piping and valve internal components has been experienced in both nuclear and fossil-fired power plants. However, with the exception of the studies published as IE Information Notices 82-22 and 82-23, neither the Staff nor anyone on its behalf has conducted any research or studies to determine the causes, effects and means of prevention, detection and mitigation of steam erosion.

The licensees of the nuclear power plants that repair components because of unanticipated events are required to document the events and describe the corrective action taken. The NRC Office of Inspection and Enforcement issued several plant specific Preliminary Notification documents prior to publication of IE Information Notices 82-22 and 82-23. All of these are available in the NRC Public Document Room. Additionally, the NRC has sponsored research concerning the detection of flaws in piping systems that was not specifically directed towards steam erosion problems. These projects have been previously reported in NUREG-0606, "Unresolved Safety Issues Summary," which also is available in the NRC Public Document Room.

INTERROGATORY NO. 9-19

It is stated in IE Information Notice 82-22 that the Oconee licensee (Duke Power Co.) theorized that reduced power operation and resultant lower quality steam contributed to accelerated steam erosion.

(a) Does the Staff accept this explanation?

(b) If not, why not?

(c) Define the term "steam quality."

(d) Explain how steam quality is related to the level of power operation.

(e) Explain how steam quality influences the degree of steam erosion.

ANSWER

(a)&(b) A safety evaluation assessing the cause of failure of the elbow at Oconee Unit 2 on June 28, 1982 was not published. However, because the impingement of entrained water at high velocity can cause erosion of metallic surfaces and operation at reduced power levels can result in lower quality steam conditions, Duke Power Company's conclusions appear to be reasonable.

(c) In light water cooled nuclear power plants, the "steam" produced is actually a steam-water mixture. The term "steam quality" is the weight of the steam divided by the total weight of the steam-water mixture. The term "per cent moisture" is the weight of the water divided by the total weight of the steam-water mixture. The sum of the "steam quality" and "per cent moisture" is one. Therefore, the lower the quality of the steam is, the higher is the per cent moisture.

(d) Generally, the turbine inlet pressure and temperature, and also the condenser pressure and temperature, are fixed. The power level is normally regulated by changing the steam mass flow rate. With the constraint on the turbine inlet and condenser conditions, the steam quality is generally lower at lower power levels.

(e) The steam quality influences the degree of steam erosion, because, the lower the quality of the steam, the higher is the amount of water present that could cause erosion.

INTERROGATORY NO. 9-20

(a) Does the Staff believe that appropriate inservice inspection and maintenance programs of licensees can detect or mitigate the effects of steam erosion?

(b) Explain the bases for the answer to (a), above.

(c) What does the Staff consider to be an "appropriate" inservice inspection or maintenance program?

(d) What measures (regulatory and enforcement) can be taken to insure that inspection and maintenance programs will be adhered to?

ANSWER

(a)&(b) The Staff believes that appropriate inservice inspection can detect the effects of steam erosion in pressure boundary piping. However, inservice inspection cannot quarantee that failures due to steam erosion will not occur. All elbows and fittings are not accessible for inservice inspection. The Staff does not have a data base to establish an appropriate examination frequency consistent with the anticipated level of degradation in order to detect the steam erosion before failure occurs.

The Staff also believes that an appropriate Inservice Testing Program (IST) can detect the effects of steam erosion on the MSIVs. 10 CFR 50.55a(g) requires that the Applicants have an acceptable IST program in accordance with the ASME Code, Section XI; a program that specifies periodic testing of pumps and valves which are important to safety.

BWR MISVs and other valves whose leakage is important for potential accident prevention or mitigation are required by the Staff to have their leaktight integrity verified periodically in accordance with Section XI of the ASME Code. However, for Containment Isolation Valves (CIVs) that are required to be leak tested in accordance with 10 CFR 50, Appendix J, the Staff accepts that testing in lieu of Section XI leak test requirements. Appendix J requires that MSIVs be leak tested at least once each 18 month period. Normally this is done by pressurizing between the two redundant MSIVs and assigning the total leakage to each valve. The acceptance criteria for that leak test is 25 SCFH per valve for the Perry Nuclear Power Plant. Valves which are maintained within these leakage limits satisfy the requirements of Section XI of the ASME Code. Such a program is capable of detecting deterioration of the seating surfaces due to various causes, including steam erosion.

(c) An appropriate inservice inspection or maintenance program is one that meets 10 CFR 50.55a(g). There presently is no requirement for the inservice inspection of non-safety related components to detect steam erosion. The Staff reviews IST programs using the guidance of Section 3.9.6, "Inservice Testing of Pumps and Valves" of the SRP (NUREG-0800). An acceptable IST program should meet the requirements of Part II of SRP Section 3.9.6, which implements 10 CFR 50.55a(g).

(d) Assurance of adherence to regulatory requirements is the function of the NRC Region III inspection staff. Region III performs inspections and audits to carry out that function.

INTERROGATORY NO. 9-21

(a) Has the Staff identified any deficiencies in the inservice inspection or maintenance programs of the licensees mentioned in IE Information Notices 82-22 and 82-23 with respect to the ability of these programs to detect or mitigate steam erosion?

(b) If so, thoroughly describe any such deficiencies.

(c) Describe any enforcement action which may have been taken against the licensees mentioned in IE Information Notices 82-22 and 82-23 (or any other licensee) as a result of steam erosion problems.

ANSWER

(a)&(b) The Staff has not identified any deficiencies in the inservice inspection or maintenance programs of licensees mentioned in IE Information Notice 82-22 with respect to the ability of these programs to detect or mitigate steam erosion because the Staff has no regulatory requirement to perform inservice inspection or maintenance of non-safety class components.

The Staff has not identified any deficiencies related to the inservice testing programs of the licensees mentioned in IE Information Notice 82-23.

(c) No enforcement actions have been taken against licensees mentioned in the IE Information Notices 82-22 or 82-23, or any other licensees, as a result of steam erosion problems.

GENERAL

INTERROGATORY NO. 9-22

For each interrogatory above, identify the person responsible for the answer, and provide his/her professional qualifications.

ANSWER

The affidavits of the persons responsible for the answers to each interrogatory and copies of their statements of professional qualifications are enclosed. The affidavits indicate the specific interrogatories to which each person, individually or jointly with others, provided answers.

INTERROGATORY NO. 9-23

Identify all documents relied upon in answering the above interrogatories, and produce all such documents not available in the NRC's Public Document Room.

ANSWER

Documents relied upon in answering the above interrogatories, if any, are identified in the answers to the interrogatories. All such documents, except SLI-8211 and SLI-8218 which have been withheld from disclosure under 10 CFR 2.790(a)(4) on the claim of their issuer that they contain "proprietary" commercial information, are available in the NRC's Public Document Room.

INTERROGATORY NO. 9-24

If there are any persons on the NRC Staff who disagree with the answers given to the above interrogatories, identify each such person and describe the nature and extent of the disagreement.

ANSWER

There are no persons on the NRC staff who disagree with the answers given to the above interrogatories.

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NUCLEAR REGULATORY COMMISSION

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

AFFIDAVIT OF JOHN O. SCHIFFGENS

I, John O. Schiffgens, being duly sworn, state as follows:

1. I am employed by the U.S. Nuclear Regulatory Commission as a Materials Engineer, Materials Engineering Branch, Division of Engineering, NRR.

2. I am the NRC staff member responsible for the answers to Interrogatories Nos. 9-1 through 9-10 of "Ohio Citizens for Responsible Energy Ninth Set of Interrogatories to NRC Staff".

3. The answers are true and complete to the best of my knowledge and belief.

John O. Schiffgens

Subscribed and sworn to before me
this [illegible] day [illegible] 1983.

Notary Public

My commission expires: [illegible]

PROFESSIONAL QUALIFICATIONS

OF

JOHN O. SCHIFFGENS

June 30, 1980 to Date	Materials Engineer Materials Engineering Branch Division of Engineering

Knowledgeable and experienced in materials and other related engineering aspects of nuclear reactors. Serves as a qualified materials engineer in the Materials Engineering Branch, Division of Engineering. Responsible for reviews, analyses, and evaluation of safety issues related to structural and mechanical components of reactor facilities licensed for power operation. Participates as a technical reviewer in evaluating applications for construction permits and operating licenses for power and non-power reactors and operational and design modifications of DOE- and DOD-owned operating facilities exempt from the licensing process.

Specific assignments include review of operating license applications for compliance with General Design Criteria 4 according to Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," and Standard Review Plan Sections 3.5.1.3, "Turbine Missiles," and 2.2.3, "Evaluation of Potential Accidents."

Education:

Bachelor of Arts, Philosophy, Saint Vincent College
Bachelor of Science, Metallurgical Engineering, University of Notre Dame
Master of Science, Nuclear Engineering, Pennsylvania State University
Doctor of Philosophy, Solid State Science, Pennsylvania State University

Experience prior to joining NRC

August 1967 to June 1974	Assistant Professor, Nuclear Engineering Department Purdue University

Taught various graduate and undergraduate courses in the Nuclear Engineering curriculum, did research in the area of irradiation effects on metals, and supervised graduate student research.

August 1974 to June 1980	Engineer, Irradiation Analysis Section Hanford Engineering Development Laboratory

Did research on the theoretical analysis of irradiation effects on metals and alloys; worked primarily in the area of computer simulation of irradiation induced atomic displacements in metals.

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AFFIDAVIT OF SUMMER B. K. SUN

I, Summer B. K. Sun, being duly sworn, state as follows:

1. I am a Nuclear Engineer in the Thermal Hyrdaulics Section, Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

2. I am the NRC Staff member responsible for the answers to Interrogatories Nos. 9-11 to 9-16 of "Ohio Citizens for Responsible Energy Ninth Set of Interrogatories to NRC Staff."

3. The answers are complete and accurate to the best of knowledge and belief.

Summer B K Sun
Summer B. K. Sun

Subscribed and sworn to be before me
this 24th day of February, 1983

Judy L. Butts
Notary Public

My commission expires: July 1, 1986

Summer B. K. Sun
Core Performance Branch
Division of Systems Integration
U.S. Nuclear Regulatory Commission

PROFESSIONAL QUALIFICATIONS

I am employed as a nuclear engineer of the Thermal-Hydraulics Section in the Core Performance Branch of the Division of Systems Integration.

I received a BS degree with Chemical Engineering Major from National Taiwan University in 1967 and a Ph.D degree with Chemical Engineering Major from University of Missouri of Columbia, Missouri, in 1974. I am a registered Professional Engineer, Certificate Number 11309, in the state of Connecticut.

In my present work assignment at the NRC, I have technical responsibility for the review of the reactor core thermal-hydraulics design submitted in BWR reactor construction permit and operating license applications. In addition, I participate in the review of analytical models used in licensing evaluation of the core thermal-hydraulic behavior under various operating and postulated accident and transient conditions. The latter responsibility includes technical review of the core spray issue and the instrumentation for monitoring inadequate core cooling to comply with the Commission requirements.

Prior to joining the NRC staff in August 1980, I was employed by Combustion Engineering Company, as a consulting engineer. I was responsible for the development and application of computer codes and methods for the analysis of transients for PWRs. My tenure at CE was from 1974 through 1980.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

(Perry Nuclear Power Plant, Units 1 and 2)

Docket No. 50-440 OL
50-441 OL

AFFIDAVIT OF MARTIN HUM

I, Martin Hum, being duly sworn, state as follows:

1. I am a Materials Engineer in the Materials Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.
2. I am the NRC Staff member jointly responsible with C.Y. Cheng, Y.C. Li and J. Page for the answers to Interrogatories 9-17, 9-18, 9-19, 9-20 and 9-21 of OCRE's Ninth Set of Interrogatories to the Staff.
3. These answers are true and complete to the best of my knowledge and belief.

Martin Hum
Martin Hum

Subscribed and sworn to before me this 25th day of February, 1983.

Patricia J. Noonan
Notary Public

My Commission expires: July 1, 1986

ATTACHMENT

PROFESSIONAL QUALIFICATIONS
OF
Martin R. Hum

EXPERIENCE:	December 1974 to Date	Materials Engineer Materials Engineering Branch Division of Engineering

I am currently a senior materials engineer in the Materials Engineering Branch, Division of Engineering. My responsibilities include the review of inspection criteria for structural and mechanical components. I have participated as a technical reviewer in evaluating nondestructive examination for applications for construction permits and operating licenses for power reactors and DOE-owned operating facilities exempt from the licensing process.

My specific asignments include review of operating license applications for compliance with Standard Review Plans for which the Inservice Inspection Section is responsible.

	June 1965 to December 1974	Mechanical Engineer U.S. Army Facilities Engineering Support Agency Fort Belvoir, Virginia

Prior to joining the NRC, I was employed as a civilian mechanical engineer with the U.S. Army Facilities Engineering Support Agency. My responsibilities included the evaluation and implementation inservice inspection for structural and mechanical components in mobile nuclear power plants. I have participated as a project engineer during the inspection, modification and/or repair of nuclear power plant systems.

EDUCATION:

Master of Science (Mechanical Engineering), George Washington University, 1971.

Bachelor of Mechanical Engineering, George Washington University, 1965.

PROFESSIONAL LICENSE:

I am registered to practice Professional Engineering in the District of Columbia, License Number 6351.

PROFESSIONAL SOCIETY MEMBERSHIP:

I am a member of the American Society of Mechanical Engineers.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of

CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

(Perry Nuclear Power Plant, Units 1 and 2)

Docket No. 50-440 OL
50-441 OL

AFFIDAVIT OF C.Y. CHENG

I, C.Y. Cheng, being duly sworn, state as follows:

1. I am a Section Leader, Inservice Inspection Section in the Materials Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.
2. I am the NRC Staff member jointly responsible with Martin Hum, Y.C. Li and Joel Page for the answers to Interrogatories 9-17, 9-18, 9-19, 9-20 and 9-21 of OCRE's Ninth Set of Interrogatories to the Staff.
3. These answers are true and complete to the best of my knowledge and belief.

C. Y. Cheng
C.Y. Cheng

Subscribed and sworn to before me this 25th day of February, 1983.

Patricia J. Nrorod
Notary Public

My Commission expires: July 1, 1986

ATTACHMENT
PROFESSIONAL QUALIFICATIONS
OF
C. Y. Cheng

EXPERIENCE:	January 1982 to Date	Section Leader Inservice Inspection Section Materials Engineering Branch Division of Engineering

As a Section Leader, I am responsible for providing technical supervision and direction to a group of materials engineers conducting reviews and evaluation of the inservice inspection aspects of the reactor coolant pressure boundary and safety-related systems as described in the applications for Construction Permits and Operating Licenses of nuclear power plants and in the proposed amendments to operating licenses.

December 1973 to December 1981	Materials Engineer/ Principal Materials Engineer Div. Tech Review/Div. Operating Reactors/ Div. Licensing, USNRC

Served as a principal reviewer for material engineering aspects of operrating reactor problems and issues related to plants under construction.

August 1967 to December 1973	Research Metallurgist Materials Science Division Argonne National Laboratory Argonne, Illinois

Performed basic and applied researches on the mechanical behavior of metals and alloys.

September 1963 to August 1967	Research Assistant/ Postdoctoral Research Metallurgist Lawrence Radiation Lab. University of California Berkeley, California

Conducted research on the mechanical metallurgy of alloys.

September 1961 to August 1963	Research Assistant Denver Research Institute University of Devener Denver, Colorado

Conducted research on the lithium - rhodium - hydrogen systems.

February 1960 to August 1961	Full time Teaching Assistant Mechanical Engineering Dep. National Taiwan University Taipei, Taiwan

Served as a full time teaching assistant for "Engineering Materials" course and its lab. Also conducted research on the mechancial properties of ferritic steels and Al-Si alloy

EDUCATION:

PhD in Engineering Science (Metallurgy), University of California, Berkeley, California,
MS in Metallurgy, University of Denver, Denver, Colorado
BS in Mechanical Engineering, National Taiwan University, Taipei, Taiwan.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.	)	Docket No. 50-440 OL
	)	50-441 OL
(Perry Nuclear Power Plant, Units 1 and 2)	)	

AFFIDAVIT OF JOEL D. PAGE

I, Joel D. Page, being duly sworn, state as follows:

1. I am a Mechanical Engineer in the Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

2. I am the NRC staff member jointly responsible with Y. Li, C. Y. Cheng and Martin Hum for the answers to Interrogatories 9-17, 9-18, 9-20 and 9-21.

3. These answers are true and complete to the best of my knowledge and belief.

Joel D. Page

Joel D. Page

Subscribed and sworn to before me
this 23rd day of February, 1983.

Patricia J. Moxon
Notary Public

My Commission expires: 7/1/86

Joel D. Page
Professional Qualifications
Mechanical Engineering Branch
Division of Engineering
U. S. Nuclear Regulatory Commission

I hold a position as a Mechanical Engineer in the above described section. I joined the Nuclear Regulatory Commission in February 1980. My responsibilities include the review of safety analysis reports and topical reports pertaining to safety related mechanical systems and components in nuclear power plants. Additionally, I am responsible for review of Inservice Testing Programs for operating plants and plants under review for operating licenses; and serve as a participating member on the ASME Working Group on Inservice Testing of Pumps and Valves.

I received a Bachelor of Science degree from Texas A&I University in 1973. In 1973 I joined the J. E. Sirrine Company, an industrial consultant, as a process (mechanical) engineer. My work included the development of process flow sheets, process design calculations, preparation of equipment specifications and bid tabulation/evaluation/recommendation.

I joined the WKM Valve Division of ACF Industries in 1975 as a product engineer. My responsibilities included design of nuclear rated gate valve components in accordance with the ASME Code, provision of technical assistance for the construction of prototype test valves, performance of seismic effect calculations and stress analyses. I utilized both in-house and commercially available finite element programming to evaluate gate valve components.

In 1978, I joined GH Bettis Corporation as Senior Engineer, Nuclear Programs. In that position, I was directly responsible for the nuclear qualifications test program, the content of the test plans, generic analyses of actuators, stress reporting and designs of actuators built primarily for nuclear service.

I am a member of the American Society of Mechanical Engineers, American Nuclear Society and I am a licensed Professional Engineer in the State of Texas.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	)	Docket No. 50-440 OL 50-441 OL

AFFIDAVIT OF YUEH LI C. LI

I, Yueh-Li C. Li, being duly sworn, state as follows:

1. I am a Mechanical Engineer in the Mechanical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

2. I am the NRC staff member jointly responsible with J. D. Page, C. Y. Cheng and Martin Hum for the answers to Interrogatories 9-17, 9-18, 9-20 and 9-21. (OCRE 9th Set Interrogatories to Staff)

3. These answers are true and complete to the best of my knowledge and belief.

Yueh-Li C. Li

Yueh-Li C. Li

Subscribed and sworn to before me this 23rd day of February, 1983.

Patricia J. Noonan

Notary Public

My Commission expires: 7/1/86

Professional Qualifications
Yueh-Li C. Li
Mechanical Engineering Branch
Division of Engineering

I am a mechanical engineer responsible for the review and evaluation of Safety Analysis Reports with respect to the mechanical engineering aspects of components, the dynamic analyses and testing of safety related systems and components and the criteria for protection against the dynamic effects associated with postulated failures of fluid systems for nuclear facilities. I am also responsible for the review and evaluation of the dynamic effects associated with the postulated rupture of piping of operating reactors for the Systematic Evaluation Program.

I received a B.S. degree in Nuclear Engineering from National Tsing-Hwa University in 1969, a M.S. in Nuclear Engineering from The Catholic University of America in 1971, and a Ph.D. in Mechanical Engineering from the same university in 1976. From 1976 to 1978, I was a senior nuclear staff engineer with Thermohydraulic Section of Nuclear Division at Bechtel Power Corporation. My work consisted of performing containment subcompartment analyses. From 1978 to 1980, I was a senior stress analyst with Plant Design Division of the same company performing piping stress analyses.

In April 1980, I joined the U.S. Nuclear Regulatory Commission as a member of the Mechanical Engineering Branch, Division of Engineering performing the type of work as previously described.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2	Docket No. 50-440 OL 50-441 OL

AFFIDAVIT OF JOHN J. STEFANO

I, John J. Stefano, being duly sworn, state as follows:

1. I am the project manager for the Perry Nuclear Power Plant, in the Division of Licensing, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission.
2. I am the NRC Staff member responsible for the answers to Interrogatories 9-22 thru 9-24 of OCRE's Ninth Set of Interrogatories to the Staff.
3. These answers are true and complete to the best of my knowledge and belief.

[signature]
John J. Stefano

Subscribed and sworn to before me this [illegible] day of February 1983.

[signature]
Notary Public

My Commission expires: [illegible] 1986

PROFESSIONAL QUALIFICATIONS

JOHN J. STEFANO

Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission

My name is John J. Stefano. I have been employed by the U. S. Nuclear Regulatory Commission (NRC) since December 1981. In February 1982, I was assigned my current duties and responsibilities as project manager for the Perry Nuclear Power Plant (PNPP), which includes the management and coordination of environmental and safety reviews documented in the Cleveland Electric Illuminating Company's (the applicant) ER and FSAR for an operating license, ensuring that all work performed by the applicant complies with all applicable NRC rules, regulations, guidelines, schedules and the provisions of AEA and NEPA. In this capacity I serve as the principle NRC point of contact and liaison between the project review staff, the applicant and other interested parties (the public, Congress, other federal, State and local governmental agencies, the media, the Advisory Committee on Reactor Safeguards, and NRC senior management).

My accomplishments to date on the PNPP project in this capacity have included: the preparation and issuance of the PNPP Safety Evaluation Report and Supplements 1 and 2 thereto (NUREG-0887); the Draft and Final Environmental Statements for the PNPP (NUREG-0884); the coordination and preparation of responses to interrogatories received in the PNPP licensing proceeding from February 1982 to the present.

I have had over 27 years of technical engineering and management experience on a wide-range of nuclear and non-nuclear programs, since having received my Bachelor of Science Degree in Aeronautical Engineering from the University of St. Louis in 1956. Post-graduate studies have included nuclear engineering and reactor safety; industrial engineering; business administration/accounting; quality assurance and mechanical engineering attending the University of Minnesota, New York University and the Carnegie-Mellon University, over the period of 1958-1975 in the pursuit of these studies.

A summary of previous positions held follows:

1977-1981	Various engineering positions with the U. S. Department of Energy involving the development and demonstration of fuel cell technology and other alternative energy technology programs; represented the Secretary of Energy on a number of national government/industry committees on energy development and economic assessment. Authored and co-authored several papers on work managed in technical journals and publications.
1965-1977	Various engineering positions with the U. S. Atomic Energy Commission involving the design, development and construction of nuclear-fueled terrestrial and space power sources and the Liquid Metal Fast Breeder reactor. Served as technical representative for AEC directors at the sites where this work was performed.

1958-1965	The position of senior project manager for the design, development, construction and test of weapon system flight simulators for the U. S. Army, Marines and Navy.
1957-1958	Active military duty with the U. S. Coast Guard
1956-1957	The position of Aeronautical engineer with the Grumman Aerospace Corporation involving the design, test and reliability analysis of flight control systems for supersonic aircraft.

UNITED STATES OF AMERICA
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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket No. 50-440 OL 50-441 OL

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF ANSWERS TO OCRE'S NINTH SET OF INTERROGATORIES TO NRC STAFF" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 1st day of March, 1983:

*Peter B. Bloch, Esq., Chairman
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

*Dr. Jerry R. Kline
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

*Mr. Glenn O. Bright
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Jay Silberg, Esq.
Shaw, Pittman, Potts and Trowbridge
1800 M Street, NW
Washington, DC 20036

Donald T. Ezzone, Esq.
Assistant Prosecuting Attorney
105 Main Street
Lake County Administration Center
Painesville, Ohio 44077

Susan Hiatt
8275 Munson Road
Mentor, Ohio 44060

Daniel D. Wilt, Esq.
P. O. Box 08159
Cleveland, Ohio 44108

Terry Lodge, Esq.
Attorney for Intervenors
915 Spitzer Building
Toledo, Ohio 43604

John G. Cardinal, Esq.
Prosecuting Attorney
Ashtabula County Courthouse
Jefferson, Ohio 44047

*Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

*Atomic Safety and Licensing
Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, DC 20555

*Docketing & Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555

James M. Cutchin IV

James M. Cutchin IV
Counsel for NRC Staff