PDR-016

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-424/425/426/427 DEC 22 1984

Laurie Fowler, Esquire
Legal Environment Assistance
Foundation
1102 Healey Building
57 Forsyth Street
Atlanta, GA 30303

IN RESPONSE REFER
TO FOIA-84-768

Dear Ms. Fowler:

This is in response to your letter dated September 26, 1984, in which you requested, pursuant to the Freedom of Information Act (FOIA), documents regarding the adequacy of the core cooling system of the Vogtle Nuclear Power Plant.

The documents listed on Appendix A are subject to your request and are enclosed.

The September 12, 1984 meeting notice is included as part of document 1, Appendix A. NRC staff have informed this office that the actual purpose of the meeting was not to discuss the "inadequate core cooling system" of the Vogtle plant. Rather, the true purpose of the meeting was to discuss core cooling instrumentation identified in the Three Mile Island (TMI) Unit 2 Task Action Plan Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling," as related to the Vogtle Plant. (See the enclosed note from Melanie A. Miller dated November 2, 1984.) The NUREG-0737 discussion of core cooling is included as document 3, Appendix A.

The documents listed on Appendix B relate tangentially to the subject of core cooling at the Vogtle plant. However, the documents are part of an on-going law enforcement investigation and are being withheld from public disclosure pursuant to Exemptions (7)(A) and (D) of the FOIA (5 U.S.C. 552(b)(7)(A) and (D)), and 10 CFR 9.5(a)(7)(i) and (iv) of the Commission s regulations. Disclosure of the information contained in this file would interfere with the investigation and would disclosure the identify of a confidential source.

Pursuant to 10 CFR 9.9 of the Commission's regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for this denial are the undersigned and Mr. James P. O'Reilly, Regional Administrator, NRC Region II.

This denial may be appealed to the NRC within 30 days from the receipt of this letter. As provided in 10 CFR 9.11, any such appeal must be in writing, addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

If you have any questions, please telephone me on 301-492-7211.

Sincerely,

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosures: As stated

FOIA-84-768

APPENDIX A

1. Extract From NUREG 0578
2. Extract From NUREG 0694
3. Extract From NUREG 0737

FOIA-84-768

APPENDIX B

Allegation Case File RII - 84-A-0145

1. Case File Cover Sheet
2. Case Chronology
3. Sign-out Sheet
4. Memo dated 09/19/84
5. Letter to Alleger - 09/29/84
6. Allegation Report RII-84-A-0145 - 09/14/84
7. Allegation Data Form

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SEP 12 1984

Docket Nos: 50-424
and 50-425

MEMORANDUM FOR: Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

FROM: Melanie Miller, Project Manager
Licensing Branch No. 4
Division of Licensing

SUBJECT: Notice of Forthcoming Meeting on Vogtle Inadequate Core Cooling

DATE and TIME: September 18, 1984
1:00 PM - 4:00 PM

LOCATION: Westinghouse Offices
4901 Fairmont Avenue
Bethesda, MD

PURPOSE: To allow the applicant the opportunity to discuss with the staff their inadequate core cooling system (NUREG-0737, Item II.F.2)

PARTICIPANTS:

NRC	Georgia Power	Westinghouse
M. Miller	J. Bailey	R. Morrison
H. Balukjian	K. Kopecky	G. Lang, et al.
T. Huang		

Melanie A. Miller

Melanie A. Miller, Project Manager
Licensing Branch No. 4
Division of Licensing

cc: See next page

~~8409210364~~

From NUREG-0578

NRR Lessons Learned Task Force
Short-Term Recommendations

TITLE: Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs (Section 2.1.3.b)

1. INTRODUCTION

General Design Criterion 13, "Instrumentation and Control," of Appendix A to 10 CFR 50, requires instrumentation to monitor variables "... for accident conditions as appropriate to assure adequate safety." In the past, GDC 13 was not interpreted to require instrumentation to directly monitor water level in the reactor vessel or the adequacy of core cooling. The instrumentation available on some operating reactors that could indicate inadequate core cooling includes core exit thermocouples, cold leg and hot leg resistance temperature detectors (RTDs), in-core neutron detectors, ex-core neutron detectors, and reactor coolant pump current. Generally, such systems were included in the reactor design to perform functions other than monitoring of core cooling or indication of vessel water level.

During the TMI-2 accident, a condition of low water level in the reactor vessel and inadequate core cooling existed and was not recognized for a long period of time. This problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, inadequate operator training, unfavorable instrument location (scattered information), and perhaps insufficient instrumentation.

The purpose of this recommendation is to provide the reactor operator with instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling.

2. DISCUSSION

With the hindsight of TMI-2, it appears that the as-designed and field-modified instrumentation at Three Mile Island Unit 2 provided sufficient information to indicate reduced reactor vessel coolant level, core voiding, and deteriorated core thermal conditions.

To preclude the failure to recognize such conditions in the future, it is appropriate to address the problem in two stages. The first is based on the detection of reduced coolant level or the existence of core voiding with the existing plant instrumentation. This would include wide range core exit thermocouples, cold leg and hot leg RTDs, coolant inventory control, in-core and ex-core detectors, vessel level (BWR), reactor coolant pump current, and other indications of coolant conditions, including coolant saturation meters (PWR). The second stage is to study and develop system modifications that would not require major structural changes to the plant and that could be implemented in a relatively rapid manner to provide more direct indication than that available with present instrumentation. These changes include PWR vessel level detectors.

A number of ideas have been discussed for the second stage by the NRC Division of Reactor Safety Research, the ACRS, and the reactor vendors. Some of the possibilities include pressure differential cells, conductivity probes, heated thermocouples, ultrasonic sounding, as well as gamma and neutron void detectors. However, we conclude that detailed engineering evaluation is required before design requirements for a direct level measurement system can be specified.

3. POSITION

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

FROM NUREG-0694

II.F.2 INADEQUATE CORE COOLING INSTRUMENTS

Develop procedures to be used by operators to recognize inadequate core cooling with currently installed instrumentation in PWRS. Install a primary coolant saturation meter. Provide a description of any additional instruments or controls needed to supplement installed equipment to provide unambiguous, easy-to-interpret indication of inadequate core cooling, procedures for use of this equipment, analyses used to develop these procedures, and a schedule for installing this equipment.

This requirement shall be met before fuel loading. See NUREG-0578, Section 2.1.3b (Ref. 4) and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24).

II.G EMERGENCY POWER FOR PRESSURIZER EQUIPMENT

Motive and control components of the power-operated relief valves and associated block valves and the pressurizer level indication shall be capable of being supplied from the offsite power source or from the emergency power buses when offsite power is not available.

This requirement shall be met before fuel loading. See NUREG-0578, Section 2.1.1 (Ref. 4), and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24).

II.K.1 IE BULLETINS ON MEASURES TO MITIGATE SMALL-BREAK LOCAs AND LOSS OF FEEDWATER ACCIDENTS

C.1.5* Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning. See Bulletins 79-06A Item 8, 79-06B Item 7, and 79-08 Item 6 in Reference 11.

C.1.10 Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability

*Table C.1 of the Action Plan lists all the requirements given in IE Bulletins.

onsite technical support center to assure that personnel in the center will not receive doses in excess of 5 rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site emergency operations facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite technical support center and direct display of radiological and meteorological parameters.

This requirement shall be met by January 1, 1981, although the safety parameter information requirements will be staged over a longer period of time. See NUREG-0578, Section 2.2.2b and 2.2.2c (Ref. 4), and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24) and April 25, 1980 (Ref. 29).

III.D.3.3 IN-PLANT RADIATION MONITORING

Provide the equipment, training, and procedures to accurately measure the radioiodine concentration in areas within the plant where plant personnel may be present during an accident.

This requirement shall be met before January 1, 1981. See NUREG-0578, Section 2.1.8c (Ref. 4), and letters of September 27 (Ref. 23) and November 9, 1979 (Ref. 24).

References

1. U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660, Vols. 1 and 2, May 1980.*

*Available for purchase from GPO sales Program, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 and National Technical Information Service, Springfield, Virginia 22161.

**Available in NRC Public Document Room for inspection and copying for a fee.

2. J. G. Kemeny, Chairman, "Report of the President's Commission on The Accident at Three Mile Island," October 1979. Available from the U.S. Government Printing Office, Washington, D.C. 20402, Attention: Superintendent of Documents, GPO Stock Number: 052-003-00718-51.

3. U.S. Nuclear Regulatory Commission, "Three Mile Island, A Report to the Commission and to the Public," USNRC Report NUREG/CR-1250, Vols. I and II, January 1980 (Vol. I) and May 1980 (Vol. II).*

4. U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," USNRC Report NUREG-0578, July 1979.*

5. U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Final Report," USNRC Report NUREG-0585, August 1979.*

6. Reports of the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation:

    a. U.S. Nuclear Regulatory Commission, "Staff Report on the Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior for Babcock & Wilcox Operating Plants," USNRC Report NUREG-0565, January 1980.

    b. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," USNRC Report NUREG-0611, January 1980.*

    c. U.S. Nuclear Regulatory Commission, "Staff Report of the Generic Assessment of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company," USNRC Report NUREG-0626, January 1980.*

    d. U.S. Nuclear Regulatory Commission, "Generic Assessment of Small-Break Loss-of-Coolant Accidents in Combustion Engineering Designed Operating Plants," USNRC Report NUREG-0635, January 1980.*

7. U.S. Nuclear Regulatory Commission, "Report of Special Review Group, Office of Inspection and Enforcement, on Lessons Learned from Three Mile Island," USNRC Report NUREG-0616, December 1979.*

8. U.S. Nuclear Regulatory Commission, "Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement," USNRC Report NUREG-0600, August 1979.*

9. U.S. Nuclear Regulatory Commission, "Report of the Siting Policy Task Force," USNRC Report NUREG-0625, August 1979.*

10. U.S. Nuclear Regulatory Commission (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," USNRC Report NUREG-0654, January 1980.*

11. U.S. Nuclear Regulatory Commission, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company," USNRC Report NUREG-0560, May 1979.*

12. Memorandum from W. J. Dircks, NRC, to Commissioners, Subject: Staff Review of the Report by the NRC Special Inquiry Group on the Accident at Three Mile Island, dated February 6, 1980.**

13. S. Chilk, U.S. Nuclear Regulatory Commission Statement of Policy on "Further Commission Guidance for Power Reactor Operating Licenses," dated June 16, 1980.**

14. Memorandum from L. V. Gossick, NRC, to Commissioners, Subject: TMI Action Plan -- Prerequirements for Resumption of Licensing, dated January 5, 1980.**

15. Memorandum from H. R. Denton, NRC, to Commissioners, Subject: Draft Action Plans for Implementing Recommendations of the President's Commission and Other Studies of TMI-2 Accident, dated December 11, 1979.**

16. Memorandum from W. J. Dirks, NRC, to NRC Office Directors, Subject: Near-Term Operating License Requirements, dated February 19, 1980.**

17. Memorandum from W. J. Dircks, NRC, to Chairman Ahearne, Subject: ACRS Report on Near-Term Operating License Requirements, dated April 1, 1980.**

18. Letter from Chairman, ACRS, to Chairman, NRC, Subject: ACRS Report on NTOL Items from Draft 3 of NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident," dated March 11, 1980.**

19. Letter from Chairman, ACRS, to Chairman, NRC, Subject: NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident, Draft 3," dated April 17, 1980.**

20. U.S. Nuclear Regulatory Commission Paper, SECY-80-230, from W. J. Dircks to Commissioners, Subject: TMI-2 Action Plan, dated May 2, 1980.**

21. Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 13, 1979.**

22. Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, Subject: Discussion of Lessons Learned Short-Term Requirements, dated October 30, 1979.**

23. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 27, 1979.**

24. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, Subject: Discussion of Lessons Learned Short-Term Requirements, dated November 9, 1979.**

25. Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, Subject: Emergency Planning, dated October 10, 1979.**

26. Letter from D. B. Vassallo, NRC, to All Pending Construction Plant Applicants, Subject: Discussion of Lessons Learned Short-Term Requirements, dated November 9, 1979.**

27. Letter from H. R. Denton, NRC, to All Power Reactor Applicants and Licensees, Subject: Qualifications of Reactor Operators, dated March 28, 1980.**

28. U.S. Nuclear Regulatory Commission, "Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities (Reprint of WASH-1293)," USNRC Report NUREG-75/111, October 1975.*

29. Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, Subject: Clarification of NRC Site Requirements for Emergency Response Facilities at Each Site, dated April 25, 1980.**

30. Letter from D. F. Ross, NRC, to All B&W Operating Plants (except TMI-1 and -2), Subject: Identification and Resolution of Long-Term Generic Issues Related to the Commission Orders of May 1979, dated August 21, 1979.**

From NUREG 0737

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Changes to Previous Requirements and Guidance

(1) Specify the "Design and Qualification Criteria" for the final ICC monitoring system in section, "Clarification" (items 7, 8, and 9), Attachment 1, and Appendix A.

(2) Specify complete documentation package to allow NRC evaluation of the final ICC monitoring systems to begin on January 1, 1981.

(3) No preimplementation review is required but postimplementation review of installation and preimplementation review before use as a basis for operator decisions are required.

(4) Installation of additional instrumentation is now required by January 1, 1982.

(5) Clarification item (6) has been expanded to provide licensees/applicants with more flexibility and diversity in meeting the requirements for determining liquid level indication by providing possible examples of alternative methods.

Previous guidance on the design and qualification criteria for upgrading of existing instrumentation was based on Regulatory Guide 1.97, which is still being developed. Detailed design requirements for incore thermocouples and additional instrumentation were not specified. The pertinent portions of draft Regulatory Guide 1.97 have now been included as Appendix A. Design requirements for incore thermocouples used in the ICC monitoring system are specified in Attachment 1. The only significant change in design requirements involves a relaxation of qualification requirements for display systems amenable to computer processing. This facilitates procurement of computer systems and makes feasible the use of cathode ray tube (CRT) displays that may be needed for proper interpretation of some reactor-water-level systems under development. This relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99% availability for the display systems, by requiring postaccident maintenance accessibility for nonredundant portions of the system, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

The staff has concluded that the previous installation requirement of January 1, 1981 for additional instrumentation is unrealistic for most licensees, due to procurement and development problems associated with proposed measurement methods. Further, the staff cannot find the proposed methods acceptable for use until development programs have been completed.

Clarification

(1) Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

(2) The evaluation is to include reactor-water-level indication.

(3) Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

(4) The indication of ICC must be unambiguous in that it should have the following properties:

    (a) It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnant boil-off); and,

    (b) It must not erroneously indicate ICC because of the presence of an unrelated phenomenon.

(5) The indication must give advanced warning of the approach of ICC.

(6) The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of ICC and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

(7) All instrumentation in the final ICC system must be evaluated for conformance to Appendix A, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

(8) If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated

hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix A, need not apply to the channel beyond the isolation device if it is designed to provide 99% availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix A, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

(9) Incore thermocouples located at the core exit or at discrete axial levels of the ICC monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.

(10) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

This requirement must be implemented by January 1, 1982.

Type of Review

A postimplementation review will be performed for installation, and a preimplementation review will be performed prior to use.

Documentation Required

By January 1, 1981, the licensee shall provide a report detailing the planned instrumentation system for monitoring of ICC. The report should contain the necessary information, either by inclusion or by reference to previous submittals including pertinent generic reports, to satisfy the requirements which follow:

(1) A description of the proposed final system including:

(a) a final design description of additional instrumentation and displays;

(b) a detailed description of existing instrumentation systems (e.g., subcooling meters and incore thermocouples), including parameter ranges and displays, which provide operating information pertinent to ICC considerations; and

(c) a description of any planned modifications to the instrumentation systems described in item 1.b above.

(2) The necessary design analysis, including evaluation of various instruments to monitor water level, and available test data to support the design described in item 1 above.

(3) A description of additional test programs to be conducted for evaluation, qualification, and calibration of additional instrumentation.

(4) An evaluation, including proposed actions, on the conformance of the ICC instrument system to this document, including Attachment 1 and Appendix A. Any deviations should be justified.

(5) A description of the computer functions associated with ICC monitoring and functional specifications for relevant software in the process computer and other pertinent calculators. The reliability of nonredundant computers used in the system should be addressed.

(6) A current schedule, including contingencies, for installation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

(7) Guidelines for use of the additional instrumentation, and analyses used to develop these procedures.

(8) A summary of key operator action instructions in the current emergency procedures for ICC and a description of how these procedures will be modified when the final monitoring system is implemented.

(9) A description and schedule commitment for any additional submittals which are needed to support the acceptability of the proposed final instrumentation system and emergency procedures for ICC.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.2, ATTACHMENT 1, DESIGN AND QUALIFICATION CRITERIA FOR PRESSURIZED-WATER REACTOR INCORE THERMOCOUPLES

(1) Thermocouples located at the core exit for each core quadrant, in conjunction with core inlet temperature data, shall be of sufficient number to provide indication of radial distribution of the coolant enthalpy (temperature) rise across representative regions of the core. Power distribution symmetry should be considered when determining the specific number and location of thermocouples to be provided for diagnosis of local core problems.

(2) There should be a primary operator display (or displays) having the capabilities which follow:

(a) A spatially oriented core map available on demand indicating the temperature or temperature difference across the core at each core exit thermocouple location.

(b) A selective reading of core exit temperature, continuous on demand, which is consistent with parameters pertinent to operator actions in connecting with plant-specific inadequate core cooling procedures. For example, the action requirement and the displayed temperature might be either the highest of all operable thermocouples or the average of five highest thermocouples.

(c) Direct readout and hard-copy capability should be available for <u>all</u> thermocouple temperatures. The range should extend from 200°F (or less) to 1800°F (or more).

(d) Trend capability showing the temperature-time history of representative core exit temperature values should be available on demand.

(e) Appropriate alarm capability should be provided consistent with operator procedure requirements.

(f) The operator-display device interface shall be human-factor designed to provide rapid access to requested displays.

(3) A backup display (or displays) should be provided with the capability for selective reading of a minimum of 16 operable thermocouples, 4 from each core quadrant, all within a time interval no greater than 6 minutes. The range should extend from 200°F (or less) to 2300°F (or more).

(4) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions.

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

(5) The instrumentation must be evaluated for conformance to Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of items 6 through 9 which follow.

(6) The primary and backup display channels should be electrically independent, energized from independent station Class IE power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. The primary display and associated hardware beyond the isolation device need not be Class IE, but should be energized from a high-reliability power source, battery backed, where momentary interruption is not tolerable. The backup display and associated hardware should be Class IE.

(7) The instrumentation should be environmentally qualified as described in Appendix B, item 1, except that seismic qualification is not required for the primary display and associated hardware beyond the isolater/input buffer at a location accessible for maintenance following an accident.

(8) The primary and backup display channels should be design to provide 99% availability for each channel with respect to functional capability to display a minimum of four thermocouples per core quadrant. The availability shall be addressed in technical specifications.

(9) The quality assurance provisions cited in Appendix B, item 5, should be applied except for the primary display and associated hardware beyond the isolation device.

November 2, 1984

Note to: Hazel Smith

From: Melanie A. Miller Ege 11/2/84

Subject. FOIA 84-768, L Fowler investigation re the adequacy of the core cooling system at Vogtle

The September 12, 1984, meeting notice upon which this request is based is included as Enclosure 1. The meeting was not to discuss the inadequacy of the core cooling system at Vogtle but rather TMI-2 Task Action Plan Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling" as indicated in the meeting notice. The NUREG-0737 discussion of this item is included as Enclosure 2.

Additionally the staff is not aware of any inadequacies with the Vogtle core cooling system. It certainly appears as if the FOIA request is the result of a misunderstanding of the meeting topic. Hopefully this clarification should take care of the requester's concerns.

Melanie A. Miller

Enclosures:
As stated

R-6002 42#