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 SUMMER B. K. SUN REGARDING ISSUE NO. 14

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

1. I am employed by the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Core Performance Branch of the Division of Systems Integration, Office of Nuclear Reactor Regulation. I wrote Section 4.4.7 of both the 1982 SER and the 1984 SSER #4 for the Perry plant. A copy of my professional qualifications is attached.

2. The purpose of this affidavit is to address Intervenor's Issue No. 14, which states:

> Applicant has not demonstrated that the Perry Nuclear Power Plant will meet regulatory safety requirements unless it installs in-core thermocouples, as suggested by staff regulatory guidelines, including Regulatory Guide 1.97, Revision 2.

The Intervenor also based this contention on NUREG-0737 requirements, studies performed by Battelle Laboratory and General Electric Company, and alleged that incore thermocouples are necessary to detect blocked flow to fuel bundles.

3. Revision 2 of Regulatory Guide 1.97 (December 1980) has a provision for installation of incore thermocouples in BWR's. (<u>See</u>: p. 8, Table 1) However, in 1981 the ACRS recommended that installation of incore thermocouple be re-evaluated. ACRS Report No. 0938, dated August 11, 1981 states at page 2:

   > 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 by reevaluated.

4. In response to the ACRS recommendation, and findings in a 1981 BWR Owners Group (BWROG) report that the effectiveness of incore thermocouples as an indication of inadequate core cooling is very limited, the staff revised the position "requiring" incore thermocouples for BWRs. Questions about the reliability of information from incore thermocouples in BWRs were raised in Section 4.4.7 of the Perry SER (May 1982) and the changed approach to ICC was described there. Subsequently, in December 1982, Supplement 1 to NUREG 0737 deleted the requirement for thermocouples for BWRs. (<u>See</u>: Section 6.1.b, p. 13) The May 1983 Revision 3 to Regulatory Guide 1.97 also eliminated this requirement. (<u>See</u> Table 2, p. 13).

5. The staff, while re-evaluating its position concerning incore thermocouples, questioned the reliability of existing water level instrumentation as the sole indication of inadequate core cooling (ICC), and requested that further study be performed by the BWROG to evaluate the need for upgrading

existing water level instrumentation to make it more reliable as an ICC detector. The staff also requested that the BWROG consider what other instrumentation might be needed in the BWR plant monitoring system.

6. The BWROG submitted two reports for staff review and approval:

   (1) SLI-8211 dated July 1982, "Review of Reactor Water Level Measurement System", which contains the BWROG's evaluation of existing water level instruments and recommendations for their improvement, and

   (2) SLI-8218 dated December 1982, "Inadequate Core Cooling Detection in BWR's", which presents an evaluation of additional instrumentation as diverse indicators of inadequate core cooling and recommendations regarding the need for such additional instrumentation (including incore thermocouples) for BWR plant monitoring systems.

7. The applicants submitted a plant specific evaluation (in letters dated January 14, 1983, November 1, 1983 and January 14, 1985) addressing the applicability of BWROG's findings (in reports SLI-8211 and SLI-8218) to Perry.

8. The results of the Staff review of Report SLI-8211 are included in the NRC generic letter 84-23 "Reactor Vessel Water Level Instrumentation in BWRs" dated October 26, 1984 where staff concurs in recommended improvements described in the report. The staff review of the applicants' letters dated January 14, 1983, November 1, 1983 and January 14, 1985 (from M. Edelman to B. Youngblood), describing modifications to the water level measurement system to make it more reliable during postulated accident conditions, is noted in the Perry SSER #4, Section 4.4.7.1. The modifications include re-routing of instrument sensing lines within the drywell

to limit the overall vertical drop to 30 inches; to reduce the differences in drops between associated reference and variable legs to be less than 26 inches, and relocation of the instrument line flow limiting orifice plates to a position near the corresponding drywell penetration. The applicants advised the Staff that, as recommended by Generic Letter 84-23, analog trip units are used at Perry rather than less reliable mechanical types, and the Perry logic design (for reactor trip and/or ESF system(s) actuation on reactor vessel low water level) has four divisions and is identical to "Plant B" in SLI-8211. In the SLI-8211 review of "Plant B", there were no cases identified which failed to provide automatic reactor trip and ECCS actuation. Therefore, based on the review of the documents described above, the staff has concluded that the Perry water level measurement system reflects the BWROG's recommendations in SLI-8211 and is sufficient to detect inadequate core cooling (ICC).

9. The staff has completed the review of SLI-8218 and agrees with its conclusion that installation of both additional ICC devices, and water level measurement reliability improvements is not justified by the resulting risk reduction. The risk remaining after inclusion of the water level measurement reliability improvements described in SLI-8211 is sufficiently small to preclude the need for further reduction in risk which would be obtained through the use of additional ICC devices analyzed by SLI-8218. Further, Appendix B to SLI-8218, as discussed in the BWROG report, calculates that the response delay time of incore 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) so that incore thermocouples will provide ambiguous information

to plant operators during the delayed response period, e.g., incore thermocouples would indicate to the operator that the core is covered while the existing water level instrumentation would indicate that the core is not covered. Therefore, the Staff agrees with the conclusion in SLI-8218 that incore thermocouples do not provide an unambiguous indication of a core uncovery condition.

10. The staff agrees with the conclusion drawn in SLI-8218 that if BWR applicants upgrade the water level system to be consistent with the recommendations cited in SLI-8211, no additional instrumentation is needed for ICC detection. Since the Perry water level instrumentation conforms with the recommendations for improvements of SLI-8211, it is the Staff's opinion that there is no additional instrumentation necessary for detection of ICC at the Perry plant.

11. As to the studies referenced by the Intervenor by Battelle Laboratory (letter from C. L. Wheeler BNL to W. V. Johnston, NRC dated April 6, 1981) and the 1981 General Electric Study, entitled "Evaluation of the Need for BWR Core Thermocouples," neither study contradicts the conclusion of the BWROG studies. A higher heat-up rate was used in the analysis provided by Battelle Laboratories. This tends to make the Battelle calculated core temperature rise more quickly after core uncovery which speeds up the thermocouples response to the 1-1½ minutes calculated by Battelle. Using the same heat-up rate for both the BWROG and Battelle analyses, the staff estimates that the calculated response delay times for both analyses are comparable. The assumed heat-up rate used in the analysis included in Appendix B to SLI-8218 is consistent

with a decay power of 2% of the initial power level, which corresponds to a time period of 700 seconds from reactor shutdown to beginning of core uncovery. The staff believes that this is a reasonable assumption to represent a power level for the core uncovery condition resulting from typical small break loss-of-coolant accident and the Battelle heat-up rate is overly conservative. The 1981 General Electric study of the need for incore thermocouples challenged by the Intervenor concluded that thermocouples would be useful for BWRs only in limited conditions, and generally, would be unnecessary. However, it was, in part, this study which raised questions as noted in the 1982 Perry SER, and led to the more complete studies by the BWROG reported in SLI-8211 and SLI-8218.

12. Regarding Intervenor's assertion of the need for incore thermocouples to detect fuel bundle blockage, the GE calculations in Appendix B to the 1981 "General Electric Evaluation of the Need for BWR Core Thermocouples" indicated that all flow paths (flow paths through the top and bottom of the fuel assembly and flow path between the fuel assembly and bypass region in the core) have to be 99 percent blocked before any fuel damage will result. Furthermore, another GE evaluation (NEDO-10174 dated October 1977) also concluded that even extensive flow blockage of a very unlikely nature would not lead to unacceptable conditions in a BWR. The staff, based on the results of the General Electric evaluation of consequences of flow blockage (NEDO-10714) and review of SLI-8218 concludes that inclusion of incore thermocouples for detection of core flow blockage is not justified by the resulting risk reduction.

13. In conclusion, for the reasons explained above, it is my opinion that the ICC detection system proposed for the Perry plant will provide an

unambiguous indication of core uncovery and that the addition of incore thermocouples would not provide significant improvement to the safety of the plant, and could create confusion about the plant condition.

14. I attest that the foregoing affidavit is true and accurate to the best of my knowledge and belief.

Summer B. K. Sun

Summer B. K. Sun

Subscribed and sworn to before me
this 28th day of January, 1985

Edythe S. Becker

Notary Public

My commission expires: 7/1/86

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 in the Core Performance Branch of the Division of Systems Integration.

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

In my present work assignment, I have technical responsibility for the review of the reactor core thermal-hydraulic design submitted in reactor construction permit and operating license applications. In addition, I participate in the review of analystical models used in licensing evaluation of the core thermal-hydraulic behavior under operation, postulated accident and transient conditions.

Prior to joining the NRC staff in August 1980, I was employed by Combustion Engineering Company (CE). I was responsible for the safety analysis method development and application of methods for the transient analyses. My responsibility at CE includes safety and performance analyses in the area of thermal-hydraulic and system designs. My tenure at CE was from 1974 through 1980.

NUREG-0737
Supplement No. 1

Clarification of TMI Action Plan Requirements

Requirements for Emergency Response Capability

Manuscript Completed: December 1982
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Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG-0737-SUP-N1

Clarification of TMI Action Plan Requirements. Supplement Number 1 Requirements for Emergency Response Capability

(U.S.) Nuclear Regulatory Commission
Washington, DC

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NUREG-0737
Supplement No. 1

Clarification of TMI Action Plan Requirements

Requirements for Emergency Response Capability

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W. Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT (200 words or less)

This document, Supplement 1 to NUREG-0737, is a letter from D. G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors, applicants for operating licenses, and holders of construction permits forwarding post-TMI requirements for emergency response capability which have been approved for implementation. On October 30, 1980, the NRC staff issued NUREG-0737, which incorporated into one document all TMI-related items approved for implementation by the Commission at that time. In this NRC report, additional clarification is provided regarding Safety Parameter Display Systems, Detailed Control Room Design Reviews, Regulatory Guide 1.97 (Revision 2) - Application to Emergency Response Facilities, Upgraded of Emergency Operating Procedures, Emergency Response Facilities, and Meteorological Data.

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

December 17, 1982

TO ALL LICENSEES OF OPERATING REACTORS, APPLICANTS FOR OPERATING LICENSES, AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: SUPPLEMENT 1 TO NUREG-0737 - REQUIREMENTS FOR EMERGENCY RESPONSE CAPABILITY (GENERIC LETTER NO. 82-33)

On October 31, 1980, the NRC staff issued NUREG-0737, which incorporated into one document all TMI-related items approved for implementation by the Commission at that time. The purpose of this letter is to provide additional clarification regarding Safety Parameter Display Systems, Detailed Control Room Design Reviews, Regulatory Guide 1.97 (Revision 2) - Application to Emergency Response Facilities, Upgrade of Emergency Operating Procedures, Emergency Response Facilities, and Meteorological Data.

The enclosures to this letter are a distillation of the basic requirements for these topics from the broad range of guidance documents that the NRC has issued (principally NUREG reports and Regulatory Guides). It is our intent that the guidance documents themselves, referred to in the enclosures, are not to be used as requirements, but rather that they are to be used as sources of guidance for NRC reviewers and licensees regarding acceptable means for meeting the basic requirements.

The following items in NUREG-0737 are affected:

I.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents

I.D.1 Control Room Design Reviews

I.D.2 Plant Safety Parameter Display Console

III.A.1.2 Upgrade Emergency Support Facilities.

III.A.2.2 Meteorological Data

The requirements and guidance contained in the enclosure to this letter replace the corresponding requirements in the affected NUREG-0737 items and should be used by you in meeting the goals of these action plan items. You should also note that the staffing levels in table 2 to the enclosure are only goals, and are not strict requirements.

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You will note that the enclosure does not specify a schedule for completing the requirements. It has become apparent, through discussions with owners' groups and individual licensees, that our previous schedules did not adequately consider the integration of these related activities. In recognition of this and the difficulty in implementing generic deadlines, the Commission has adopted a plan to establish realistic plant-specific schedules that take into account the unique aspects of the work at each plant. By this plan, each licensee is to develop and submit its own plant-specific schedule which will be reviewed by the assigned NRC Project Manager. The NRC Project Manager and licensee will reach an agreement on the final schedule and in this manner provide for prompt implementation of these important improvements while optimizing the use of utility and NRC resources.

Applicants for construction permits are expected to comply with the requirements of 10 CFR 50.34(f), and should consider this document to be additional guidance in meeting these requirements. For holders of construction permits and applicants for operating licenses, plant-specific schedules for the implementation of these requirements will be developed in a manner similar to that being used for operating reactors, taking into consideration the degree of completion of the power plant.

In order to answer questions you may have regarding the Commission's policy on these issues and the implementation process to be used by project managers, regional workshops will be conducted by senior staff members according to the following schedule:

Region I	Washington, D. C.	- Week of 2/14/83
Region II	Atlanta, Ga.	- Week of 2/21/83
Region III	Chicago, Ill.	- Week of 2/21/83
Region IV & V	San Francisco, CA	- Week of 2/28/83

You will be notified of specific locations and times for the workshops at a later time.

Accordingly, pursuant to 50.54(f), operating reactor licensees and holders of construction permits are requested to furnish, no later than April 15, 1983 a proposed schedule for completing each of the basic requirements for the items identified in the enclosures to this letter. You are encouraged to work closely with your NRC Project Manager during this process so that we can reach an agreement on the final schedule as quickly as possible. In addition, you are requested to submit with it a description of your plans for phased implementation and integration of the emergency response activities. Your plans for integration will be reviewed as part of our evaluation of your proposed schedule. After the staff completes this evaluation, it will take action, as necessary, to assure that such requirements and commitments are appropriately enforceable.

V

This request for information was approved by the Office of Management and Budget under clearance number 3150-0065 which expires May 31, 1983. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D. C. 20503.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
Supplement to NUREG-0737

CONTENTS

CONTENTS (Continued)

EMERGENCY RESPONSE CAPABILITY

1. INTRODUCTION

This supplement was prepared as a result of a review by the Committee to Review Generic Requirements (CRGR). The supplement represents the staff's attempt to distill the fundamental requirements for nuclear plant Emergency Response Capability from the wide range of guidance documents that the NRC has issued. It is not intended that these guidance documents (NUREG reports and Regulatory Guides) be implemented as written; rather, they should be regarded as useful sources of guidance for licensees and NRC staff regarding acceptable means for meeting the fundamental requirements contained in this document. It is also not intended that either the guidance documents or the fundamental requirements are to be considered binding legal requirements at this time. As indicated below, however, the fundamental requirements will be translated into binding legal requirements in the manner specified.

These requirements are a further delineation of the general guidance issued previously by the Commission in its regulations, orders and policy statements on emergency planning and TMI issues. It is intended that these requirements would be applicable to licensees of operating nuclear power plants. For applicants for a construction permit (CP) or manufacturing license (ML), the requirements described in this document must be supplemented with the specific provisions in the rule specifying licensing requirements for pending CP and ML applications. Thus, compliance with requirements in this document may not be sufficient to meet the related requirements in 10 CFR 50.34(f) and Appendix E. In this regard, it is expected that the staff would review CP and ML applications against the guidance in the current Standard Review Plan (which includes the provisions of NUREG-0718) and this might lead to more detailed requirements than prescribed in this document in order to satisfy the requirements of 50.34(f) and Appendix E.

Based on discussions with licensees, the staff has learned that many of the Commission approved schedules for emergency response facilities probably will not be met. In recognition of this fact and the difficulty of implementing generic deadlines, plant-specific schedules will be established which take into account the unique status of each plant. The following sequence for developing implementation schedules will be used.

The requirements for emergency response capabilities and facilities are being transmitted to licensees by this supplement and are being promulgated to NRC staff. The letter which forwards this supplement requests that licensees submit a proposed schedule for completing actions to comply with the requirements.

Each licensee's proposed schedule will then be reviewed by the assigned NRC Project Manager, who will discuss the subject with the licensee and mutually agree on schedules and completion dates. The implementation dates will then be formalized into an enforceable document.

The requirements in this document do not alter previously issued guidance, which remains in effect. This document does attempt to place that guidance in perspective by identifying the elements that the NRC staff believes to be essential to upgrade emergency response capabilities. The proposal to formalize implementation dates in an enforceable document reflects the level of importance which the NRC staff attributes to these requirements. The Commission does not believe that existing guidance should be imposed in this manner, but rather that it be used as guidance to be considered in upgrading emergency response capabilities. This indicates the distinction which the staff believes should be made between the requirements and guidance.

The following sections describe the requirements, their interrelationhips, and NRC actions to improve management of emergency response regulations. Reference documents are cited with a description of content as it relates to specific initiatives.

The requirements set forth in this document have been reviewed by the Commission and, at a meeting held July 16, 1982, were approved by the Commission as appropriately clarifying and providing greater detail with respect to related TMI Action Plan requirements contained in NUREG-0737 for all operating license applicants. These requirements are, therefore, to be accorded the status of approved NUREG-0737 items as set forth in the Commission's "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses" (45 FR 85236), December 24, 1980). In this connection, the provisions for scheduling set forth herein supersede any schedules with respect to such items contained in NUREG-0737. Accordingly, the requirements should be used by the staff and by adjudicatory boards as appropriate clarifications and interpretation of the related NUREG-0737 items.

The requirements set forth in this document are believed to be consistent with the requirements regarding related items for construction permits and manufacturing licenses contained in 10 CFR 50.34(f) and 10 CFR Part 50, Appendix E. Accordingly, no changes to these regulations are required.

2. USE OF EXISTING DOCUMENTATION

The following NUREG documents are intended to be used as sources of guidance and information, and the Regulatory Guides are to be considered as guidance or as an acceptable approach to meeting formal requirements. The items by virtue of their inclusion in these documents shall not be misconstrued as requirements to be levied on licensees or as inflexible criteria to be used by NRC staff reviewers.

NUREG Report		Titles
0696	-	Functional Criteria for Emergency Response Facilities
0700	-	Guidelines for Control Room Design Reviews
0799	-	Draft Criteria for Preparation of Emergency Operating Procedures (to be superseded by NUREG-0899)
0801	-	Evaluation Criteria for Detailed Control Room Design Reviews
0814	-	Methodology for Evaluation of Emergency Response Facilities
0818	-	Emergency Action Levels for Light Water Reactors
0835	-	Human Factors Acceptance Criteria for SPDS
0899	-	Guidelines for the Preparation of Emergency Operating Procedures: Resolution of Comments on NUREG-0799

Regulatory Guides		Titles
1.23 (Rev. 1)	-	Meteorological Measurement Program for Nuclear Power Plants
1.97 (Rev. 2)	-	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident
1.101 (Rev. 2)	-	Emergency Planning for Nuclear Power Plants
1.47	-	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

3. COORDINATION AND INTEGRATION OF INITIATIVES

3.1 The design of the Safety Parameter Display System (SPDS), design of instrument displays based on Regulatory Guide 1.97 guidance, control room design review, development of function oriented emergency operating procedures, and operating staff training should be integrated with respect to the overall enhancement of operator ability to comprehend plant conditions and cope with emergencies. Assessment of information needs and display formats and locations should be performed by individual licensees. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS may obviate the need for large-scale control room modifications. Installation of the SPDS should not be delayed by slower progress on other initiatives, and should not be contingent on completion of the control room design review. Nor should other initiatives, such as upgraded emergency operating procedures, be impacted by delays in SPDS procurement. While the NRC does not plan to impose additional requirements on licensees regarding SPDS, the NRC will work with the industry to assure the development of appropriate industry standards for SPDS systems.

3.2 Implementation of part or all of Regulatory Guide 1.97 (Rev. 2) represents a control room improvement. The implementation of control room improvements is not contingent on implementing Technical Support Center (TSC) and Emergency Operations Facility (EOF) requirements.

3.3 The Technical Support Center (TSC) and Emergency Operations Facility (EOF) are dependent on control room improvements in terms of communication and instrumentation needs among the TSC, EOF, and control room. TSC and EOF facilities are not necessarily dependent on each other. The Operational Support Center (OSC) is independent of TSC and EOF.

3.4 The three groups of initiatives--SPDS, control room improvements, and emergency response facilities (TSC, EOF, OSC)-- have the following interrelationships:

a. The SPDS is an improvement because it enhances operator ability to comprehend plant conditions and interact in situations that require human intervention. The SPDS could affect other control room improvements that licensees may consider. In some cases, a good SPDS could obviate the need for extensive modifications to control rooms.

b. New instrumentation that may be added to the control room should be considered a requirement for inclusion in the design of the TSC and EOF only to the extent that such instrumentation is essential to the performance of TSC and EOF functions.

c. The SPDS and control room improvements are essential elements in operator training programs and the upgraded plant-specific emergency operating procedures.

d. Acquisition, processing, and management of data for SPDS, control room improvements, and emergency response facilities should be coordinated.

3.5 Specific implementation plans and reasonable, achievable schedules for improvements that will satisfy the requirements will be established by agreement between the NRC Project Manager and each individual licensee. The NRC office responsible for implementing each requirement will develop procedures identifying the following.

a. The respective roles of NRR, IE, and Regional Offices in managing implementation, checking licensee rate of progress, and verifying compliance, including the extent to which NRC review and inspection is necessary during implementation.

b. Procedural methods and enforcement measures that could be used to ensure NRC staff and licensee attention to meeting mutually agreed upon schedules without significant delays and extensions.

3.6 The NRC Project Manager for each nuclear power plant is assigned program management responsibility for NRC staff actions associated with implementing emergency response initiatives. The NRC Project Manager is the principal contact for the licensee regarding these initiatives.

3.7 The NRC will make allowances for work already done by licensees in a good-faith effort to meet requirements as they understand them. For each case in which a licensee would have to remove or rip out emergency response facilities or equipment that was installed in good faith to meet previous guidance in order to meet the basic requirements described in this document, the Director of the Office of Nuclear Reactor Regulation or Inspection and Enforcement will review the circumstances and determine whether removal is necessary or existing facilities or equipment represent an acceptable alternative. Any regulatory position that would require the removal or major modification of existing emergency response facilities or equipment requires the specific approval of the responsible Office Director.

3.8 The NRC recognizes that acceptable alternative methods of phasing and integrating emergency response activities may be developed. Each licensee needs flexibility in integrating these activities, taking into account the varying degree to which the licensee has implemented past requirements and guidance. An example of a way in which these activities could be integrated is discussed below. Other methods of integration proposed by licensees would be reviewed considering licensees' progress on each initiative.

a. SPDS

(1) Review the functions of the nuclear power plant operating staff that are necessary to recognize and cope with rare events that (a) pose significant contributions to risk, (b) could cause operators to make cognitive errors in diagnosing them, and (c) are not included in routine operator training programs.

(2) Combine the results of this review with accepted human factors principles to select parameters, data display, and functions to be incorporated in the SPDS.

(3) Design, build, and install the SPDS in the control room and train its users.

b. To be done in parallel without delaying SPDS, complete emergency operating procedure technical guidelines that will be used to develop plant-specific emergency operating procedures.

c. Using these EOP technical guidelines, the SPDS design, and accepted human factors principles, conduct a review of the control room design. Apply the results of this review to:

(1) Verify SPDS parameter selection, data display, and functions.

(2) Develop plant-specific EOPs.

(3) Design control room modifications that correct conditions adverse to safety (reduce significant contributions to risk), and add additional instrumentation that may be necessary to implement Regulatory Guide 1.97.

(4) Train and qualify plant operating staff regarding upgraded EOPs and modifications.

d. Verify, prior to finalization of designs for modifications and of procedures and training, that the functions of control room operators in emergencies can be accomplished (i.e., that the individual initiatives have been integrated sufficiently to meet the needs of control room operators and provide adequate emergency response capabilities).

e. Implement EOPs and install control room modifications coincident with scheduled outages as necessary, and train operators in advance of these changes as they are phased into operation.

4. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

4.1 Requirements

a. The SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

b. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.

c. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) provides the operators with the information necessary for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, 4 in Appendix A; 10 CFR Part 100; single failure requirements). The SPDS need not meet requirements of the single-failure criteria and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. Procedures which describe the timely and correct safety status assessment when the SPDS is and is not available, will be developed by the licensee in parallel with the SPDS. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.

d. There is a wide range of useful information that can be provided by various systems. This information is reflected in such staff documents as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97

Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information that should be provided for a particular plant shall be based on engineering judgement of individual plant licensees, taking into account the importance of prompt implementation.

e. The SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily preceived and comprehended by SPDS users.

f. The minimum information to be provided shall be sufficient to provide information to plant operators about:

   (i) Reactivity control

   (ii) Reactor core cooling and heat removal from the primary system

   (iii) Reactor coolant system integrity

   (iv) Radioactivity control

   (v) Containment conditions

   The specific paramenters to be displayed shall be determined by the licensee.

4.2 Documentation and NRC Review

a. The licensee shall prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Such analysis, along with the specific implementation plan for SPDS shall be reviewed as described below.

b. The licensee's proposed implementation of an SPDS system shall be reviewed in accordance with the licensee's technical specifications to determined whether the changes involve an unreviewed safety question or change of technical specifications. If they do, the shall be processed in the normal fashion with prior NRC review. If the changes do not involve an unreviewed safety question or a change in the technical specifications, the licensee may implement such changes without prior approval by NRC or may request a pre-implementation review and approval. If the changes are to be implemented without prior NRC approval, the licensee's analysis shall be submitted to NRC promptly on completion of review by the licensee's offsite safety review committee. Based on the results of NRC review, the Director of IE or the Director of NRR may request or direct the licensee to cease implementation if a serious safety question is posed by the licensee's proposed system, or if the licensee's analysis is seriously inadequate.

4.3 Integration

Prompt implementation of an SPDS is a design goal and of primary importance. The schedule for implementing SPDS should not be impacted by schedules for the control room design review and development of function-oriented emergency operating procedures. For this reason, licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to the other initiatives. If reasonable, this schedule will be accepted by NRC.

4.4 Reference Documents

NUREG-0660	--	Need for SPDS identified
NUREG-0737	--	Specified SPDS
NUREG-0696	--	Functional Criteria for SPDS
NUREG-0835	--	Specific acceptance criteria keyed to NUREG-0696
Reg. Guide 1.97 (Rev. 2)	--	Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

5. DETAILED CONTROL ROOM DESIGN REVIEW

5.1 Requirements

a. The objective of the control room design review is to "improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (from NUREG-0660, Item I.D.1). As a complement to improvements of plant operating staff capabilities in response to transients and other abnormal conditions that will result from implementation of the SPDS and from upgraded emergency operating procedures, this design review will identify any modifications of control room configurations that would contribute to a significant reduction of risk and enhancement in the safety of operation. Decisions to modify the control room would include consideration of long-term risk reduction and any potential temporary decline in safety after modifications resulting from the need to relearn maintenance and operating procedures. This should be carefully reviewed by persons competent in human factors engineering and risk analysis.

b. Conduct a control room design review to identify human engineering discrepancies. The review shall consist of:

(i) The establishment of a qualified multidisciplinary review team and a review program incorporating accepted human engineering principles.

(ii) The use of function and task analysis (that had been used as the basis for developing emergency operating procedures Technical Guidelines and plant specific emergency operating procedures) to identify control room operator tasks and information and control requirements during emergency operations. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters.

(iii) A comparison of the display and control requirements with a control room inventory to identify missing displays and controls.

(iv) A control room survey to identify deviations from accepted human factors principles. This survey will include, among other things, an assessment of the control room layout, the usefulness of audible and visual alarm systmes, the information recording and recall capability, and the control room environment.

c. Assess which human engineering discrepancies are significant and should be corrected. Select design improvements that will correct those discrepancies. Improvements that can be accomplished with an enhancement program (paint-tape-label) should be done promptly.

d. Verify that each selected design improvement will provide the necessary correction, and can be introduced in the control room without creating any unacceptable human engineering discrepancies because of significant contribution to increased risk, unreviewed safety questions, or situations in which a temporary reduction in safety could occur. Improvements that are introduced should be coordinated with changes resulting from other improvement programs such as SPDS, operator training, new instrumentation (Reg. Guide 1.97, Rev. 2), and upgraded emergency operating procedures.

5.2 Documentation and NRC Review

a. All licensees shall submit a program plan within two months of the start of the control room review that describes how items 1, 2 and 3 above will be accomplished. The staff will review the program plans as licensees conduct their reviews, and selected licensee will undergo an in-progress audit by the NRR human factors staff based on the program plans and advice from resident inspectors and Project Managers.

b. All licensees shall submit a summary report of the completed review outlining proposed control room changes, including their proposed schedules for implementation. The report will also provide a summary justification for human engineering discrepancies with safety significance to be left uncorrected or partially corrected.

c. The staff will review the summary reports, and within two weeks after receipt of the licensee's summary report, will inform licensees whether a pre-implementation onsite audit will be conducted. The decision will be based on the content of the program plan, the summary report, and the results of NRR in-progress audits, if any. The licensee selection for pre-implementation audit may or may not include licensees selected for in-progress audits under paragraph 1.

d. For control rooms selected for pre-implementation onsite audit, within one month after receipt of the summary report, the NRC will conduct:

(i) A pre-implementation audit of proposed modifications (e.g., equipment additions, deletions and relocations, and proposed modifications).

(ii) An audit of the justification for those human engineering discrepancies of safety significance to be left uncorrected or only partially corrected.

The audit will consist of a review of the licensee's record of the control room reviews, discussions with the licensee review team, and usually a control room visit. Within a month after this onsite audit, NRC will issue its safety evaluation report (SER).

e. For control rooms for which NRC does not perform a pre-implementation onsite audit, NRC will conduct a review and issue its SER within two months after receipt of the licensee's summary report. The review shall be similar to that conducted for pre-implementation plants under paragraph 4 above, except that it does not include a specific audit. The SER shall indicate whether, based on the review carried out, changes in the licensee's modification plan are needed to assure operational safety. Flexibility is considered in the control room review, because certain control board discrepancies can be overcome by techniques not involving control board changes. These techniques could include improved procedures, improved training, or the SPDS.

f. The following approach will be used for OL review. For OL applications with SSER dates prior to June 1983, licensing may be based on either a Preliminary Design Assessment or a Control Room Design Review (CRDR) at the applicant's option. However, applicants who choose the Preliminary Design Assessment option are required to perform a CRDR after licensing. For applications with SSER dated after June 1983, Control Room Design Review will be required prior to licensing.

g. After the staff has issued an SER and licensees have addressed any open issues, they may begin their upgrade according to an approved schedule that has been negotiated with the staff.

5.3 Reference Documents

NUREG-0585	--	States that licensees should conduct review.
NUREG-0660 (Rev. 1)	--	States that NRR will require reviews for operating reactors and operating licensee applicants.
NUREG-0700	--	Final guidelines for CRDR.
NUREG-0737	--	States that requirement was issued June, 1980, final guidance not yet issued.
NUREG-0801	--	Staff evaluation criteria.

6. REGULATORY GUIDE 1.97 - APPLICATION TO EMERGENCY RESPONSE FACILITIES

6.1 Requirements

a. Functional Statement

Regulatory Guide 1.97 provides data to assist control room operators in preventing and mitigating the consequences of reactor accidents.

b. Control Room

Provide measurements and indication of Type A, B, C, D, E variables listed in Regulatory Guide 1.97 (Rev. 2). Individual licensees may take exceptions based on plant-specific design features. BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements. It is acceptable to rely on currently installed equipment if it will measure over the range indicated in Regulatory Guide 1.97 (Rev. 2), even if the equipment is presently not environmentally qualified. Eventually, all the equipment required to monitor the course of an accident would be environmentally qualified in accordance with the pending Commission rule on environmental qualification.

Provide reliable indication of the meteorological variables (wind direction, wind speed, and atmospheric stability) specified in Regulatory Guide 1.97 (Rev. 2) for site meteorology. No changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity (up to about 10 miles) of the plant site. Information on meteorological conditions for the region in which the site is located shall be available via communication with the National Weather Service. These requirements supersede the clarification of NUREG-0737, Item III.A.2.2.

c. Technical Support Center (TSC)

The Type A, B, C, D and E variables that are essential for performance of TSC functions shall be available in the TSC.

(i) BWR incore thermocouples and continuous offsite dose monitors are not required pending their further development and consideration as requirements.

(ii) The indicators and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

d. Emergency Operations Facility (EOF)

(i) Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be available in the EOF.

(ii) The EOF data indications and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

6.2 Documentation and NRC Review

NRC review is not a prerequisite for implementation. Staff review will be in the form of an audit that will include a review of the licensee's method of implementing Regulatory Guide 1.97 (Rev. 2) guidance and the licensee's supporting technical justification of any proposed alternatives.

The licensee shall submit a report describing how it meets these requirements. The submittal should include documentation which may be in the form of a table that includes the following information for each Type A, B, C, D, E variable shown in Regulatory Guide 1.97 (Rev. 2).

(a) instrument range

(b) environmental qualification (as stipulated in guide or state criteria)

(c) seismic qualification (as stipulated in guide or state criteria)

(d) quality assurance (as stipulated in guide or state criteria)

(e) redundance and sensor(s) location(s)

(f) power supply (e.g., Class 1E, non-Class 1E, battery backed)

(g) location of display (e.g., control room board, SPDS, chemical laboratory)

(h) schedule (for installation or upgrade)

Deviations from the guidance in Regulatory Guide 1.97 (Rev. 2) should be explicitly shown, and supporting justification or alternatives should be presented.

7. UPGRADE EMERGENCY OPERATING PROCEDURES (EOPs)

7.1 Requirements

a. The use of human factored, function oriented, emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors, without the need to diagnose specific events.

b. In accordance with NUREG-0737, Item I.C.1, reanalyze transients and accidents and prepare Technical Guidelines. These analyses will identify operator tasks, and information and control needs. The analyses also serve as the basis for integrating upgraded emergency operating procedures and the control room design review and verifying the SPDS design.

c. Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide.

d. Provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs.

e. Implement upgraded EOPs.

7.2 Documentation and NRC Review

a. Submit Technical Guidelines to NRC for review. NRC will perform a pre-implementation review of the Technical Guidelines. Within two months of receipt of the Technical Guidelines, NRC will advise the licensees of their acceptability.

b. Each licensee shall submit to NRC a procedures generation package at least three months prior to the date it plans to begin formal operator training on the upgraded procedures. NRC approval of the submittal is not necessary prior to upgrading and implementing the EOPs. The procedures generation package shall include:

(i) Plant-Specific Technical Guidelines -- plant-specific guidelines for plants not using generic technical guidelines. For plants using generic technical guidelines, a description of the planned method for developing plant specific EOPs from the generic guidelines, including plant specific information.

(ii) A Writer's Guide that details the specific methods to be used by the licensee in preparing EOPs based on the Technical Guidelines.

(iii) A description of the program for validation of EOPs.

(iv) A brief description of the taining program for the upgraded EOPs.

c. All procedures generation packages will be reviewed by the staff. On an audit basis for selected facilities, upgraded EOPs will be reviewed. The details and extent of this review will be based on the quality of the procedures generation packages submitted to NRC. A sampling of upgraded EOPs will be reviewed for technical adequacy in conjunction with the NRC Reactor Inspection Program.

7.3 Reference Documents

NUREG-0600,
Item I.C.1, I.C.8, I.C.9

NUREG-0799 -- (Superseded by NUREG-0899)

8. EMERGENCY RESPONSE FACILITIES

8.1 Regulations

10 CFR 50.47(b)(6) (for Operating License applicants) -- Requirement for prompt communications among principal response organizations and to emergency personnel and to the public.

10 CFR 50.47(b)(8) -- Requirement for emergency facilities and equipment to support emergency response.

10 CFR 50.47(b)(9) -- Requirement that [illegible]te methods, systems and equipment for assessing and monitoring [illegible]ual or potential offsite consequences of a radiological emergency condition are in use.

10 CFR 50.54(q) (for Operating Reactors) -- Same requirement as 10 CFR 50.47(b) plus 10 CFR 50, Appendix E.

10 CFR 50, Appendix E, Paragraph IV.E
Requirement for:

"1. Equipment at the site for personnel monitoring;"

"2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;"

"3. Facilities and supplies at the site for decontamination of onsite individuals;"

"4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;"

"5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on site;"

"6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;"

"7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;"

"8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;"

"9. At least one onsite and one offsite communications system; each system shall have a backup power source."

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

"a. Provision for communications with contiguous State/local governments within the plume exposure pathway (emergency planning zone) EPZ. Such communications shall be tested monthly."

"b. Provisions for communication with Federal emergency response organizations. Such communication systems shall be tested annually."

"c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually."

"d. Provisions for communication by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly."

Within this section on emergency response facilities, the Technical Support Center (TSC), Operational Support Center (OSC) and Emergency Operations Facility (EOF) are addressed separately in terms of their functional statements and recommended requirements. The subsections on Documentation and NRC Review and Reference Documents that follow the EOF discussion apply to this entire section on emergency response facilities.

8.2 Technical Support Center (TSC)

8.2.1 Requirements

a. The TSC is the onsite technical support center for emergency response. When activated, the TSC is staffed by predesignated technical, engineering, senior management, and other licensee personnel, and five pre-designated NRC personnel. During periods of activation, the TSC will operate uninterrupted to provide plant management and technical support to plant operations personnel, and to relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations. The TSC will perform EOF functions for the Alert Emergency class and for the Site Area Emergency class and General Emergency class until the EOF is functional.

The TSC will be:

b. Located within the site protected area so as to facilitate necessary interaction with control room, OSC, EOF and other personnel involved with the emergency.

c. Sufficient to accommodate and support NRC and licensee predesignated personnel, equipment and documentation in the center.

d. Structurally built in accordance with the Uniform Building Code.

e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

f. Provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

g. Provided with reliable voice and data communications with the control room and EOF and reliable voice communications with the OSC, NRC Operations Centers and state and local operations centers.

h. Capable of reliable data collection, storage, analysis, display and communication sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

(i) the variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2) that are essential for performance of TSC functions; and

(ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and National Weather Service data available by voice communication for the region in which the plant is located.

Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

i. Provided with accurate, complete and current plant records (drawings, schematic diagrams, etc.) essential for evaluation of the plant under accident conditions.

j. Staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and be fully operational within approximately 1 hour after activation.

k. Designed taking into account good human factors engineering principles.

8.3 Operational Support Center (OSC)

8.3.1 Requirements

a. When activated, the OSC will be the onsite area separate from the control room where predesignated operations support personnel will assemble. A predesignated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by control room, TSC and EOF personnel.

The OSC will be:

b. Located onsite to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

c. Capable of reliable voice communications with the control room, TSC and EOF.

8.4 Emergency Operations Facility (EOF)

8.4.1 Requirements

a. The EOF is a licensee controlled and operated facility. The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment, development of recommendations for public protective actions, and coordination of emergency response activities with Federal, State and local agencies.

When the EOF is activated, it will be staffed by predesignated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radio-active releases from the plant and to determine dose projections.

The EOF will be:

b. Located and provided with radiation protection features as described in Table 1 (previous guidance approved by the Commission) and with appropriate radiological monitoring systems.

c. Sufficient to accommodate and support Federal, State, local and licensee predesignated personnel, equipment and documentation in the EOF.

d. Structurally built in accordance with the Uniform Building Code.

e. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

f. Provided with reliable voice and data communications facilities to the TSC and control room, and reliable voice communication facilities to OSC and to NRC, State and local emergency operations centers.

g. Capable of reliable collection, storage, analysis, display and communication of information on containment conditions, radiological releases and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF functions shall be available in the EOF:

   (i) variables from the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Rev. 2), and

   (ii) the meteorological variables in Regulatory Guide 1.97 (Rev. 2) for site vicinity and regional data available via communication from the National Weather Service.

h. Provided with up to date plant records (drawings, schematic diagrams, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

i. Staffed using Table 2 (previous guidance approved by the Commission) as a goal. Reasonable exceptions to goals for the number of additional staff personnel and response times for their arrival should be justified and will be considered by NRC staff.

j. Provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

k. Designed taking into account good human factors engineering principles.

8.4.2 <u>Documentation and NRC Review</u>

The conceptual design for emergency response facilities (TSC, OSC, and EOF) have been submitted to NRC for review. In many cases, the lack of detail in these submittals has precluded an NRC decision of acceptability. Some designs have been disapproved because they clearly did not meet the intent of the applicable regulations. NRC does not intend to approve each design prior to implementation, but rather has provided in this document those requirements which should be satisfied. These requirements provided a degree of flexibility within which licensees can exercise management prerogatives in designing and building emergency response facilities (ERF) that satisfy specific needs of each licensee. The foremost consideration regarding ERFs is that they provide adequate

adequate capabilities of licensees to respond to emergencies. NUREG guidance on ERFs has been intended to address specific issues which the Commission believes should be considered in achieving improved capabilities.

Licensees should assure that the design of ERFs satisfies these requirements. Exemptions from or alternative methods of implementing these requirements should be discussed with NRC staff and in some cases could require Commission approval. Licensees should continue work on ERFs to complete them according to schedules that will be negotiated on a plant-specific basis. NRC will conduct appraisals of completed facilities to verify that these requirements have been satisfied and that ERFs are capable of performing their intended functions. Licensees need not document their actions on each specific item contained in NUREG-0696 or 0814.

8.4.3 Reference Documents (Emergency Response Facilities)

10 CFR 50.47(b) -- Requirements for emergency facilities and equipment for OLs.

10 CFR 50.54(q) and Appendix E, Paragraph IV.E -- Requirements for emergency facilities and equipment for ORs.

NUREG-0660 -- Description of and implementation schedule for TSC, OSC and EOF.

Eisenhut letter to power reactor licensees 9/13/79 -- Request for commitment to meet requirements

Denton letter to power reactor licensees 10/30/79 -- Clarification of requirements.

NUREG-0654 -- Radiological Emergency Response Plans

NUREG-0696 -- Functional criteria for emergency response facilities.

NUREG-0737 -- Guidance on meteorological monitoring and dose assessment.

Eisenhut letter to power reactor license 2/18/81 -- Commission approved guidance on location, habitability and staff for emergency facilities. Request and deadline for submittal of conceptual design of facilities.

NUREG-0814 (Draft Report for Comment) -- Methodology for evaluation of emergency response facilities.

NUREG-0818 (Draft Report for Comment) -- Emergency Action Levels

Reg. Guide 1.97 (Rev. 2) -- Guidance for variables to be used in selected emergency response facilities.

COMJA-80-37, January 21, 1981 -- Commission approval guidance on EOF location and habitability.

Secretary memorandum S81-19, February 19, 1981 -- Commission approval of NUREG-0696 as general guidance only.

TABLE 1

EMERGENCY OPERATIONS FACILITY

Option 1 Two Facilities	Option 2 One Facility
Close-in Primary: Reduce Habitability* o within 10 miles o protection factor = 5 o ventilation isolation with HEPA (no charcoal)	o At or Beyond 10 miles. o No special protection factor. o If beyond 20 miles, specific approval required by the Commission, and some provision for NRC site team closer to site. o Strongly recommended location be coordinated with offsite authorities.
Backup EOF o between 10-20 miles o no separate, dedicated facility o arrangements for portable backup equipment o strongly recommended location be coordinated with offsite authorities o continuity of dose projection and decision making capability	

For both Options:

- located outside security boundary
- space fo- about 10 NRC employees
- none designated for severe phenomena, e.g., earthquakes

*Habitability requirements are only for the part of the EOF in which dose assessments communications and decision making take place.

If a utility has begun construction of a new building for an EOF that is located with 5 miles, that new facility is acceptable (with less than protection factor of 5 and ventilation isolation and HEPA) provided that a backup EOF similar to "B" in Option 1 is provided.

TABLE 2

MINIMUM STAFFING REQUIREMENTS FOR NRC LICENSEES
FOR NUCLEAR POWER PLANT EMERGENCIES

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions: On Shift*	30 min.	60 min.
Plant Operations and Assessment of Operational Aspects		Shift supervisor (SRO)	1	--	--
		Shift foreman (SRO)	1	--	--
		Control-room operators	2	--	--
		Auxiliary operators	2		
Emergency Direction and Control (Emergency Coordinator)***		Shift technical advisor, shift supervisor, or designated facility manager	1**	--	--
Notification/ Communication****	Nofity licensee, state local, and federal personnel & maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency operations facility (EOF) director	Senior manager	--	--	1
	Offsite dose assessment	Senior health physics (HP) expertise	--	1	--
	Offsite surveys		--	2	2
	Onsite (out-of-plant)		--	1	1
	Inplant surveys	HP technicians	1	1	1
	Chemistry/radio-chemistry	Rad/chem technicians	1	--	1

NOTE: Source of this table is NUREG-0654, "Functional Criteria for Emergency Response Facilities."

TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions: On Shift*	30 min.	60 min.
Plant System Engineering, Repair and Corrective Actions	Technical support	Shift technical advisory	1	--	--
		Core/thermal hydraulics	--	1	--
		Electrical	--	--	1
		Mechanical	--	--	1
	Repair and corrective actions	Mechanical maintenance/	1**	--	1
		Radwaste operator			1
		Electrical maintenance/	1**	1	1
		instrument and control		1	--
		(I&C) technician	--	1	--
Protective Actions (In-Plant)	Radiation protection: a. Access control b. HP Coverage for repair, corrective actions, search and rescue first-aid, & firefighting c. Personnel monitoring d. Dosimetry	HP technicians	2**	2	2
Firefighting	--	--	Fire brigade per technical specification	Local support	
Rescue Operations and First-Aid	--	--	2**	Local support	

TABLE 2 (Continued)

Major Functional Area	Major Tasks	Position Title or Expertise	Capability for Additions On Shift*	30 min.	60 min.
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security personnel	All per security plan		
		Total	10	11	15

*For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control-room operator, and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.

**May be provided by shift personnel assigned other functions.

***Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Directo of minute-to-minute facility operations remains with senior manager in technical support center or control room.

****May be performed by engineering aide to shift supervisor.

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

October 26, 1984

TO ALL BOILING WATER REACTOR (BWR) LICENSEES OF OPERATING REACTORS
(EXCEPT LACROSSE, BIG ROCK POINT, HUMBOLDT BAY AND DRESDEN-1)

Gentlemen:

SUBJECT: REACTOR VESSEL WATER LEVEL INSTRUMENTATION IN BWRs
(GENERIC LETTER NO. 84-23)

The water level instrumentation in a BWR is relied upon for controlling feedwater, actuating emergency systems, and for providing the operators information which is used as basis for actions to assure adequate core cooling. Many of the actions in the emergency procedures guidelines are keyed to reactor water level.

The NRC staff has reviewed the S. Levy, Inc. report SLI-8211, "Review of BWR Reactor Vessel Water Level Measurement System," which was commissioned by the BWR Owners Group. The report provides a generic evaluation of water level instrumentation adequacy of BWR/2 through BWR/6 plants and identifies several improvements in BWR water level measurement and instrumentation systems which will improve the reliability and accuracy of those systems. The staff has concluded that changes identified in the emergency procedure guidelines are adequate for the short term, but permanent physical improvements should be made on a deliberate schedule to reduce the burden on the operator. Three potential improvement categories are presented below:

- Improvements to plant(s) that will reduce level indication errors caused by high drywell temperature. These improvements include prevention of reference leg overheating or reduction of the vertical drops in the drywell. (Vertical drop should be measured from the condensation pot to the drywell exit point. Maximum drop would allow an indicated level at the bottom of the normal operating range when actual level is just above lower tap for worst flashing condition.) Those plants for which the vertical drop in the drywell has already been minimized will not have to make additional changes for the drywell heating effect.

- Review of plant experience relating to mechanical level indication equipment. Plant experience shows mechanical level equipment is more vulnerable to failure or malfunction than analog equipment. A number of plants have already connected analog trip units to their level transmitters to improve reliability and accuracy. Those plants that use mechanical level indication should replace the mechanical level indication equipment with analog level transmitters unless operating experience confirms high reliability.

- Changes to the protection system logic that may be needed for those plants in which operator action may be required to mitigate the consequences of a break in a reference leg and a single failure in a protection system channel associated with an intact reference leg. Changes will generally result in additional transmitters to satisfy the single failure criterion. This improvement, under evaluation by NRC, may be needed in plants where an analysis has demonstrated a vulnerability.

Implementation of the first two categories of improvements will give increased assurance that the level instrumentation will provide the inadequate core cooling instrumentation required by NUREG-0737, Item II.F.2 and thereby safisfy this requirement. Please submit within 30 days a description of your plans to implement these improvements and your proposed schedule.

The last improvement is still being evaluated by the staff; hence it is not a requirement at this time. However, should the continuing evaluation show that there is a significantly high priority. It will be identified as a new generic letter.

This request for information was approved by the Office of Management and Budget under clearance number 3150-0065 which expires September 30, 1985.

Sincerely,

Frank J. Miraglia

for

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation