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

Reprinted February 1989

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

June 15, 1982

TO ALL LICENSEES OF OPERATING PLANTS, APPLICANTS FOR AN OPERATING LICENSE, AND HOLDERS OF CONSTRUCTION PERMITS

Gentlemen:

SUBJECT: NUCLEAR POWER PLANT STAFF WORKING HOURS
(Generic Letter No. 82-12)

On February 8, 1982, we forwarded to you a copy of a policy statement issued by the Commission entitled "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors." That policy statement has now been slightly revised and the revised version is attached to this letter. The forwarding letter (Generic Letter 82-02) stated that actions were underway to incorporate the Commission policy on working hours into Regulatory Guide 1.33 and into NUREG-0737 (Item I.A.1.3).

Publication of Regulatory Guide 1.33 is now anticipated in June 1982, following release of ANS 3.2. Revised pages of NUREG-0737 that incorporate the Commission policy on working hours are enclosed.

Our letter of February 8, 1982 requested that you take action as necessary to revise the administrative section of your technical specifications to assure that your plant administrative procedures follow the revised working hour guidelines, including a provision for documentation of authorized deviations which should be available for NRC review. You should review your past actions to assure that they are consistent with the attached revised policy statement. Note that the revised guidelines are to be incorporated by October 1, 1982.

Sincerely,

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

Attachment:
Revision to Item I.A.1.3
of NUREG-0737

I.A.1.3 SHIFT MANNING

Pages I.A.1.3-1 through I.A.1.3-3 are replaced in their entirety by the following. A bar in the margin indicates changes from the material originally included on these pages.

Position

This position defines shift manning requirements for normal operation. The letter of July 31, 1980 from D. G. Eisenhut to all power reactor licensees and applicants (copy attached) sets forth the interim criteria for shift staffing (to be effective pending general criteria that will be the subject of future rulemaking). Overtime restrictions were also included in the July 31, 1980 letter.

Changes to Previous Requirements and Guidance

Errors were discovered in the last column of the table attached to the letter of July 31, 1980. A corrected table is enclosed; a bar in the margin indicates the correction. (See p. I.A.1.3-4.)

The overtime requirements have been rewritten to incorporate the provisions of a Commission policy statement on working hours and operator fatigue.

Clarification

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to prevent situations where fatigue could reduce the ability of operating personnel to keep the reactor in a safe condition. The controls established should assure that, to the extent practicable, personnel are not assigned to shift duties while in a fatigued condition that could significantly reduce their mental alertness or their decision making ability. The controls shall apply to the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours", dated February 1, 1980 (copy attached) discusses the concern of overtime work for members of the plant staff who perform safety-related functions. The guidance contained in IE Circular No. 80-02 was amended by the July 31, 1980 letter. In turn, the overtime guidance of the July 31, 1980 letter was revised in Section I.A.1.3 of NURG-0737. The NRC has issued a policy statement (attached) which further revises the overtime guidance as stated in NUREG-0737. This guidance is as follows:

Enough plant operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime. The objective is to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period (all excluding shift turnover time).

c. A break of at least eight hours should be allowed between work periods (including shift turnover time).

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on shift.

Recognizing that very unusual circumstances may arise requiring deviation from the above guidelines, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management. The paramount consideration in such authorization shall be that significant reductions in the effectiveness of operating personnel would be highly unlikely. Authorized deviations to the working hour guidelines shall be documented and available for NRC review.

In addition, procedures are encouraged that would allow licensed operators at the controls to be periodically relieved and assigned to other duties away from the control board during their tours of duty.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning Oct. 1, 1982.

See Section III.A.1.2 for minimum staffing and augment capabilities for emergencies.

Applicability

This requirement applies to all licensees of operating reactors and applicants for operating licenses.

Implementation

(1) Overtime administrative procedures shall be established for operating reactors by Oct. 1, 1982 and by fuel loading for applicants for operating license.

(2) Staffing requirements shall be completed by July 1, 1982 for operating reactors and by fuel load for operating license applicants.

Type of Review

A post-implementation review will be performed on operating reactors. Applicants for operating licenses will be reviewed prior to implementation.

Documentation Required

The documentation required is as noted in the letter of July 31, 1980.

Technical Specification Changes Required

Changes to Technical Specifications will be required for overtime administrative procedure and staffing requirements.

References

NUREG-0660

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," February 1, 1980

Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, July 31, 1980

Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors,

POLICY ON FACTORS CAUSING FATIGUE OF OPERATING PERSONNEL AT NUCLEAR REACTORS

Licensees of operating plants and applicants for operating licenses shall establish controls to prevent situations where fatigue could reduce the ability of operating personnel to keep the reactor in a safe condition. The controls should focus on shift staffing and the use of overtime--key job-related factors that influence fatigue.

The objective of the controls would be to assure that, to the extent practicable, personnel are not assigned to shift duties while in a fatigued condition that could significantly reduce their mental alertness or their decision making capability. The controls shall apply to the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel).

Enough plant operating personnel should be employed to maintain adequate shift coverage without routine heavy use of overtime. The objective is to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period (all excluding shift turnover time).

c. A break of at least eight hours should be allowed between work periods (including shift turnover time).

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Recognizing that very unusual circumstances may arise requiring deviation from the above guidelines, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management. The paramount consideration in such authorization shall be that significant reductions in the effectiveness of operating personnel would be highly unlikely.

In addition, procedures are encouraged that would allow licensed operators at the controls to be periodically relieved and assigned to other duties away from the control board during their tour of duty.

NUREG-0737
Supplement No. 1

Clarification of TMI Action Plan Requirements

Requirements for Emergency Response Capability

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Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

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, Upgrade of Emergency Operating Procedures, Emergency Response Facilities, and Meteorological Data.

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.

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

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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 licensee 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 adequate methods, systems and equipment for assessing and monitoring actual 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 Documentation and NRC Review

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 for about 10 NRC employees
- none designated for severe phenomena, e.g., earthquakes

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

f a utility has begun construction of a new building for an EOF that is located with 5 miles, that new acility is acceptable (with less than protection factor of 5 and ventilation isolation and HEPA) provided hat 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:	HP technicians	2**	2	2
	a. Access control				
	b. HP Coverage for repair, corrective actions, search and rescue first-aid, & firefighting				
	c. Personnel monitoring				
	d. Dosimetry				
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.

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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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